TVA

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May 25, 2001

10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - INFORMATION RELATED TO SQN TRITIUM PROGRAM

As discussed with NRC staff on April 4, 2001, TVA is providing the enclosed information in advance of submitting a license amendment to address TVA's proposed irradiation of Tritium Producing Burnable Absorber Rods (TPBARs) for the Department of Energy (DOE) at SQN (Tritium Program). This information will allow NRC to become more familiar with the nature and the scope of the proposed Tritium Program in advance of TVA's license amendment request. TVA plans to periodically update the NRC Project Manager as to the expected date for submission of the license amendment.

Enclosure 1 provides Framatome-Advanced Nuclear Power (ANP) Report BAW-10237, which addresses items such as responses to plant specific interface items, core design, plant specific confirmatory checks, TVA program changes, and TPBAR design and fabrication. Please note Section 1.5, page 1-5 of Enclosure 1 provides references as to where each of the 17 plant specific interface items are addressed.

Enclosure 2 provides a discussion regarding TPBAR consolidation and the addition of a TPBAR consolidation fixture.

D030

Enclosure 3 provides a discussion regarding methodology changes for the spent fuel pool cooling analysis.

As you are aware, DOE submitted a classified/proprietary version (NDP-98-153, Revision 1) and an unclassified/non-proprietary version (NDP-98-181, Revision 1) of the Tritium Production Core (TPC) Topical Report for NRC review. Both versions of the TPC Topical Report have been used in providing the enclosed information. In order to maintain this information in an unclassified form, any classified text, tables, and figures that will be affected by the plant-specific application of TPBARs have been omitted. Copies of the classified documents are available for NRC review at the Pacific Northwest National Laboratory offices.

As you are also aware, NRC reviewed the TPC Topical Report and issued NUREG-1672, "Safety Evaluation Report (SER) Related to the Department of Energy's Topical Report on the Tritium Production Core," documenting its review. TVA used these documents as references and will include the appropriate plant-specific evaluations and analyses, including the 17 interface items listed in NUREG-1672, Section 5.1, in its license amendment request.

There are no new regulatory commitments in this submittal. In accordance with NRC RIS 2001-05, only one paper copy of this document is being sent to the NRC Document Control Desk. If you have any questions about this submittal, please contact me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

Framatome ANP
BAW-10237

Implementation and Utilization of
Tritium Producing Burnable Absorber Rods (TPBARS)
in Sequoyah Units 1 and 2

BAW-10237
May 2001

IMPLEMENTATION AND UTILIZATION OF TRITIUM PRODUCING BURNABLE ABSORBER RODS (TPBARS) IN SEQUOYAH UNITS 1 AND 2

(Unclassified, Non-Proprietary Version)

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TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Section **Page**

LIST OF TABLES

LIST OF FIGURES

LIST OF ACRONYMS

AFD	Axial Flux Difference
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
ANL-W	Argonne National Laboratory - West
AOA	Axial Offset Anomaly
ARI	All Rods In
ARO	All Rods Out
ART	Adjusted Reference Temperature
ASL	Acceptable Suppliers List
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
BOL	Beginning of Life
BP	Burnable Poison
BPRA	Burnable Poison Rod Assembly
CCS	Component Cooling System
CFR	Code of Federal Regulations
Ci	Curie
CHF	Critical Heat Flux
CLWR	Commercial Light Water Reactor
COMS	Cold Overpressure Mitigation System
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
CZP	Cold Zero Power
DAC	Derived Air Concentration
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
DOPC	Doppler Only Power Coefficient
DWMS	Demineralized Water Makeup System
EC	Equilibrium Cycle
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EFPY	Effective Full Power Years

LIST OF ACRONYMS

EOL	End of Life
EQ	Environmental Qualification
ERCW	Essential Raw Water Cooling
ERG	Emergency Response Guideline
ESFAS	Engineered Safety Features Actuation System
FC	First Cycle
FRA-ANP	Framatome ANP
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GDT	Gas Decay Tank
GVR	Gas Volume Ratio
HHSI	High Head Safety Injection
HTC	Heat Transfer Coefficient
HFP	Hot Full Power
HZP	Hot Zero Power
ID	Inner Diameter
IFBA	Integral Fuel Burnable Absorber
INEEL	Idaho National Engineering and Environmental Laboratory
LAR	License Amendment Request
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LTOP(S)	Low Temperature, Overpressure Protection (System)
M&E	Mass and Energy
M&TE	Measurement and Test Equipment
MOL	Middle of Life
MPH	Material Properties Handbook
MTC	Moderator Temperature Coefficient
MWd/mtU	Megawatt Days per Metric Ton of Uranium
MW_t	Megawatts Thermal
NDE	Nondestructive Examination
NPZ	Nickel-Plated Zirconium
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OD	Outer Diameter

LIST OF ACRONYMS

ODCM	Offsite Dose Calculation Manual
PASF	Post Accident Sampling Facility
PCT	Peak Cladding Temperature
PNNL	Pacific Northwest National Laboratory
ppm	Parts Per Million
P/T	Pressure-Temperature
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RAOC	Relaxed Axial Offset Control
RBPVS	Reactor Building Purge Ventilation System
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
REP	Radiological Emergency Preparedness Program
RHR(S)	Residual Heat Removal (System)
RIL	Rod Insertion Limit
RTP	Rated Thermal Power
RV	Reactor Vessel
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
scf	Standard Cubic Feet
SDM	Shutdown Margin
SER	Safety Evaluation Report
SFP	Spent Fuel Pool or Pit
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SG	Steam Generator
SQN	Sequoyah Nuclear Plant
SQNREF	Sequoyah Reference Core Design (non-TPBAR)
SQNTPC	Sequoyah Tritium Production Core Design
SRP	Standard Review Plan
SS	Stainless Steel
SSWS	Station Service Water System
STP	Standard Temperature and Pressure
TCF	TPBAR Consolidation Fixture

LIST OF ACRONYMS

TEDE	Total Effective Dose Equivalent
TPBAR	Tritium Production Burnable Absorber Rod
TPC	Tritium Production Core
TPCRD	Tritium Production Core Reference Design
TPCTR	Tritium Production Core Topical Report
TS	Technical Specification
TVA	Tennessee Valley Authority
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
VCT	Volume Control Tank
WABA	Wet Annular Burnable Absorber
WBN	Watts Bar Nuclear Plant
w/o	Weight Percent or Without (depending on context)
WOG	Westinghouse Owners Group

SECTION 1 INTRODUCTION

1.1 PURPOSE OF PROGRAM

The U.S. Department of Energy (DOE) is planning to produce tritium for the National Security Stockpile by irradiating Tritium Producing Burnable Absorber Rods (TPBARs) in a number of commercial light water reactors (CLWRs). The Tennessee Valley Authority's (TVA) Sequoyah Nuclear Plant (SQN) and Watts Bar Nuclear Plant (WBN) have been selected by the DOE to accomplish this mission.

A topical report (Reference 1) was written that addressed the safety and licensing issues associated with incorporating a full complement of TPBARs in a CLWR, specifically a pressurized water reactor (PWR). The U.S. Nuclear Regulatory Commission's (NRC) Standard Review Plan (SRP) (Reference 2) was used as the basis for evaluating the impact of the TPBARs on a reference plant. The NRC reviewed Reference 1 and issued a Safety Evaluation Report (SER) (Reference 3) to support plant specific licensing of TPBARs in a PWR. A number of issues were cited in References 1 and 3 requiring the performance of plant specific evaluations and analyses to demonstrate that no significant safety issues are raised by the operation of a PWR with a full complement of TPBARs.

1.2 DESCRIPTION OF EFFORT

References 1 and 3 defined the plant specific evaluations and analyses required for SQN. Evaluations and analyses associated with radiation calculations are being reviewed and verified. This information will be provided later. Therefore, there are portions of items 1-7 which are noted as to be provided later. Specifically, the scope of work concentrated on:

1. Addressing the 17 plant specific interface issues listed in NUREG-1672, Section 5.1. The following interface items have been submitted previously under a separate cover letter:
   a. LOCTAJR
   b. Anticipated Transients Without Scram (ATWS)

   Items 1.a and 1.b have been approved and closed in SERs dated January 17, 2001 and March 16, 2001 respectively.
2. Identifying and evaluating the significant differences as they apply to SQN relative to the Tritium Production Core Topical Report (TPCTR).
3. Providing confirmation of no adverse impact for the plant specific confirmatory checks required by the TPC topical report.
4. Providing evaluations of plant specific confirmatory checks that revealed an impact by TPBARs on reactor performance, plant systems, and plant operations.
5. Addressing plant specific changes consisting of:
   a. Required Technical Specification (TS) changes for implementation and utilization of TPBARs at SQN.
   b. SQN thermal power up-rate of 1.3%. The uprate is not required for the implementation and utilization of TPBARs, however, analyses and evaluations performed for this report assumed up-rated thermal power conditions because TVA anticipates implementation of this uprate prior to initial insertion of TPBARs into SQN.
6. Addressing other items cited in the SER, e.g.,
   a. TPBAR surveillance program.
   b. Lead Test Assembly (LTA) post irradiation results.
7. Providing additional information regarding the behavior of failed TPBARs during normal operation and during a LBLOCA.

1.3 SEQUOYAH PLANT PARAMETERS

The TVA Sequoyah Units 1 and 2 are Westinghouse designed 4-loop pressurized water reactors with a rated thermal power of 3411 MW_t. Each unit contains 193 fuel assemblies of the 17x17 design. A fuel assembly consists of 264 fuel rods, 24 guide thimbles, and one instrumentation tube. Excess reactivity is typically controlled using 53 Ag-In-Cd rod cluster control assemblies (RCCA), burnable poison rod assemblies (BPRA), integral burnable absorbers (gadolinium oxide dispersed in UO_2 fuel rods), and soluble boron in the reactor coolant system (RCS).

The preceding discussion provides a brief description of the Reference Sequoyah Reactor. Throughout this report, the following terms and acronyms will be used to distinguish a tritium production reactor from a reference reactor:

> Sequoyah reference reactor or plant (SQNREF) - The current Sequoyah reactor or plant rated at 3411 MW_t that has no TPBARs and therefore does not purposely produce tritium.
>
> Sequoyah tritium production reactor or plant (SQNTPC) - The Sequoyah reactor or plant rated at 3455 MWt with a core designed to produce tritium using a complement of TPBARs. TVA anticipates implementation of a 1.3% thermal power uprate to 3455 MW_t prior to initial insertion of the TPBARs in Units 1 and/or 2.
>
> Tritium production reactor reference design (TPCRD) - The reference reactor or plant described in the Topical Report (Reference 1) with a core designed to produce tritium using a complement of TPBARs.

Table 1-1 provides a comparison of Nuclear Steam Supply System (NSSS) parameters and features for the TPCRD, SQNREF, and SQNTPC. The TPCRD was used as the basis for the reference TPBAR studies described in Reference 1. It was assumed that the TPCRD was representative of candidate plants for the CLWR tritium program. SQNTPC was used as the basis for all evaluations and analyses described in this report.

Various key core design parameters are compared in Table 1-2 for the TPCRD and SQNTPC. TPBARs will be inserted into the guide thimble locations of selected fuel assemblies at Sequoyah to meet tritium production requirements. The exceptions will be assemblies that are located under RCCAs or contain BPRAs, source rods, and/or thimble plugs. Table 1-3 shows various key physical parameters for SQNTPC.

The parameters provided in this section are primarily NSSS performance parameters. Other Sequoyah specific parameters (e.g., core peaking factors, core by-pass flow, etc.) are presented in Sections 2 and 3, which describe the evaluations and analyses performed to demonstrate the feasibility of TPBAR use in Sequoyah.

1.4 APPLICATION OF TRITIUM PRODUCTION CORE (TPC) TOPICAL REPORT TO SEQUOYAH

This report utilizes the TPC Topical Report (TPCTR) (Reference 1) and Reference 3 (SER) as the bases for the plant specific evaluations and analyses performed for Sequoyah. Extensive analyses, testing, and evaluations of TPBARs and their impact on a CLWR incorporating TPBARs were documented in the TPCTR. It is the intent of this report not to reproduce the evaluations presented in TPCTR that showed no impact of TPBAR utilization in a CLWR. However, each Standard Review Plan section in the TPCTR was reviewed to determine whether the "no impact" conclusion was valid for Sequoyah. Plant specific evaluations (and analyses if required) were performed for Sequoyah as recommended in the TPCTR. Some of these remain to be confirmed until the independent review of the TVA radiation calculations have been completed. This information will be submitted later.

1.4.1 Sequoyah Report Sections Referencing the TPC Topical Report

Table 1-4 is intended as a guide that cites the specific section used to evaluate the impact of TPBARs on Sequoyah. Each SRP item (designated in Table 1-4 by "SRP Section Number", "SRP Section Title", and "NDP-98-181, Revision 1 Section") evaluated in Reference 1 is listed in Table 1-4. If the specific item was not impacted by the incorporation of TPBARs in the TPCRD and Sequoyah, the fourth column (entitled "Plant Specific Evaluation Needed") will contain a "No" for that item. If the specific item was impacted by the incorporation of TPBARs in the TPCRD and/or in Sequoyah, then a "Yes" will be shown in the fourth column to denote that a specific evaluation was required. Column five (entitled "Sequoyah Report Section") will contain the appropriate section number where the Sequoyah specific evaluation is discussed. When the fifth column of Table 1-4 contains an "NA" for a specific item, then the evaluation performed in Reference 1 (see Column 3) has been determined to be applicable to SQNTPC.

It should also be noted that the numbering convention used in this report is identical to Reference 1 down to the third level (e.g. Section 1.4.2). Sections 1 and 4 are the exception to this convention. Sections that appear to be missing have been purposely omitted because either the information contained in the TPCTR is applicable to SQNTPC, the item for Sequoyah is addressed in Section 1.5 as an interface issue, or the specific evaluation of the item is presented in Section 4, Table 4-1.

1.4.2 Identification of Differences

A review of the TPCTR and the SER was completed to identify any differences that exist between SQNTPC and the TPCRD. In addition, the review included identifying any differences between the NRC conclusions documented in the SER and SQNTPC. The noted differences are discussed in each section of this report as appropriate. There are differences which will be supplied later and are so noted. As part of the review, new information was identified concerning TPBAR performance following failures during normal plant operation and post-LBLOCA. This information is further discussed in Section 3.0.

1.5 SEQUOYAH PLANT SPECIFIC INTERFACE ISSUES

During its review of the TPCTR, the NRC determined there are certain plant specific interface issues for which the licensee must submit additional information and analyses. This information would be used to support a plant specific license amendment to the facility's operating license for authorization to operate a tritium production core. Each specific interface issue has been evaluated for SQN and is discussed below. As cited in Sections 1.5.16 and 1.5.17, submittals to the NRC have been made to address these items.

Note that references cited by each specific interface issue will be contained within the individual interface issue section.

The following is a listing of the NUREG-1672 interface items along with section number where these items are addressed in this report:

1. Handling of TPBARs (1.5.1)
2. Procurement and Fabrication Issues (1.5.2)
3. Compliance with DNB Criterion (1.5.3)
4. Reactor Vessel Integrity Analysis (Appendices G and H to 10 CFR Part 50 and 10 CFR 50.61) (1.5.4)
5. Control Room Habitability Systems (1.5.5)
6. Specific Assessment of Hydrogen Source and Timing or Recombiner Operation (1.5.6)
7. Light-Load Handling System (1.5.7)
8. Station Service Water System (1.5.8)
9. Ultimate Heat Sink (1.5.9)
10. New and Spent Fuel Storage (1.5.10)
11. Spent Fuel Pool Cooling and Cleanup System (1.5.11)
12. Component Cooling Water System (1.5.12)
13. Demineralized Water Makeup System (1.5.13)
14. Liquid Waste Management System (1.5.14)
15. Process and Effluent Radiological Monitoring and Sampling System (1.5.15)
16. Use of LOCTA_JR Code for LOCA analyses (1.5.16)
17. ATWS Analysis (1.5.17)

1.5.1 Handling of TPBARs

Action

NUREG-1672, Section 1.3, "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."

NUREG-1672, Section 2.9.2, "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."

NUREG-1672, Section 3.7, "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."

Response

TPBAR handling during the consolidation and shipping phase of the program was not discussed in the above SER sections and was so noted.

TVA has completed a preliminary design of a TPBAR Consolidation Fixture (TCF) to be installed in the cask loading pit for consolidation activities (see Figures 1.5.1-1 and 1.5.1-2). The TCF is quality related in accordance with TVA's NRC accepted QA Program. It will normally be stored in the cask lay-down area when not in use. The TCF fixture includes a video monitoring system, lighting, and tools designed to remove TPBARs from its baseplate. The TPBARs are deposited into a consolidation canister (up to 300 TPBARs per canister). The loaded canister is transferred back into the spent fuel pool for short term storage until ultimately being placed into shipping casks for transport off-site to DOE.

The TPBAR consolidation canister loading concept has been successfully demonstrated at DOE's Savannah River Site facility. The completed consolidation fixture and tools will be tested prior to shipment and also after installation to verify proper operation prior to actual use.

Consolidation Sequence

Each tritium core is loaded with certain fuel assemblies containing up to 24 TPBARs (multiples of 4) attached to a baseplate (TPBAR assembly). The TPBARs then undergo an irradiation cycle. After the core is unloaded to the spent fuel pool during refueling, the irradiated TPBAR assemblies are removed from the fuel and transferred to available storage locations within the spent fuel pool using the burnable poison rod assembly tool. Material accountability for TPBAR assemblies is administratively controlled.

TPBARs are normally shipped with the new fuel assemblies to the reactor site. TPBAR assemblies that are inserted into once burned fuel are transferred from their storage location into the required fuel assemblies using a burnable poison rod assembly tool. Approximately 30 days after refueling is complete, TPBAR consolidation begins.

The canisters (see Figure 1.5.1-3) that receive the irradiated TPBARs are transferred into the spent fuel pool and placed into the consolidation fixture when required. A TPBAR assembly is then withdrawn from its available storage location and moved from the spent fuel pool to the consolidation fixture using the TPBAR assembly handling tool suspended from the SFP Bridge crane. A TPBAR release tool is then utilized by personnel on the platform to detach individual TPBARs from the baseplate. The TPBAR slides along frame guides, through a funnel and into a roller brake, to limit its velocity, and then into the consolidation canister. The funnel, roller brake assembly, and canister are angled at approximately 15° to enable the TPBARs to stack efficiently into the canister to maximize the loading. All activities take place underwater at a safe shielding water depth.

After TPBARs have been removed from a baseplate, the baseplate and any attached thimble plugs will be removed from the fixture (utilizing a hand held baseplate tool or a TPBAR assembly handling tool suspended from the SFP Bridge crane), and the baseplate and thimble plugs placed in storage. The process is repeated until the canister is filled with up to 300 TPBARs. Disposal or storage of the baseplates and thimble plugs will be in accordance with accepted radwaste programs.

The loaded canister is removed and transported to a designated storage position in the spent fuel pool storage rack using the canister handling tool suspended from the SFP Bridge crane. The next empty consolidation canister is placed into the consolidation fixture and the process is repeated until all TPBARs irradiated during the fuel cycle have been consolidated. The consolidation fixture is then removed from the cask load pit, and stored in the cask lay-down area.

Subsequently, a shipping cask is placed into the cask loading pit. The cask is handled by the Auxiliary Building crane in accordance with NUREG-0612 program requirements. The canisters are transferred into the submerged cask. The cask is removed from the cask loading pit, drained of water and decontaminated, packaged and certified for shipment. This shipping process is repeated until all TPBARs irradiated during the past operating cycle have been shipped.

1.5.2 Procurement and Fabrication Issues

Action

NUREG-1672, Section 1.3, "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."

NUREG-1672, Section 2.17.1, "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."

Response

The Department of Energy (DOE) procures TPBAR design, fabrication, irradiation, and transportation services for the delivery of irradiated TPBARs to the DOE Tritium Extraction Facility. The major DOE suppliers are PNNL, WesDyne, TVA, and a yet to be determined supplier for irradiated TPBAR Transportation Services.

The Pacific Northwest National Laboratory (PNNL) in Richland, Washington developed and qualified the design and fabrication processes, fabricated and delivered TPBARs for use as lead test assemblies (LTAs), obtained lead test assembly irradiation services from TVA, and performed LTA TPBAR post irradiation examinations. In addition, PNNL's scope includes design and fabrication process improvements associated with supporting full scale tritium production, material and subcomponent procurements in sufficient initial quantities to support commencement of TPBAR irradiation under a full scale production program, and transition of TPBAR designer of record responsibilities to WesDyne International LLC (WesDyne). WesDyne is a wholly owned subsidiary of the Westinghouse Electric Company LLC that operates under a separate Board Of Directors. WesDyne uses the Westinghouse Quality Management System (QMS).

The WesDyne TPBAR Fabrication Facility, located at the Westinghouse Fuel Fabrication Plant in Columbia South Carolina will receive materials and subcomponents purchased by PNNL; procure materials and services, assemble, process, and fabricate final TPBARs; and deliver certified TPBARs to TVA or TVA's nuclear fuel manufacturers for use in TVA reactor cores. In addition, WesDyne will assume long term designer of record responsibilities from PNNL in support of the full scale tritium production program.

Upon receipt of certified TPBARs, TVA's fuel vendor will install TPBARs onto baseplates in accordance with their respective NRC accepted QA Program.

TVA will irradiate the DOE furnished TPBARs. After irradiation, TVA will consolidate TPBARs and prepare them for DOE shipments to the Tritium Extraction Facility.

The activities associated with TPBAR design, material and service procurements, fabrication, and delivery are being performed under the auspices of TVA's NRC Accepted QA program (TVA-NQA-PLN89A). Refer to Section 2.17 for further details.

TVA is responsible for obtaining safety-related components and services from TVA accepted suppliers. DOE is managing the overall Tritium Production Program including issuance of major procurements. TVA requires that all safety-related materials, items, and services be procured from TVA accepted suppliers and comply with TVA specified technical, functional, and quality requirements. In order to ensure that the DOE documents used to obtain safety-related materials, items, and services adequately address the TVA requirements, TVA reviews applicable DOE documents for acceptance.

TVA evaluates PNNL and WesDyne for TPBAR design, material and service procurements, fabrication and assembly, and delivery and places them on TVA's Acceptable Suppliers List (ASL). TVA maintains a list of acceptable suppliers in accordance with TVA's NRC accepted QA program. Maintenance of suppliers on TVA's ASL includes annual evaluations, audits, and surveillance of selected supplier activities.

In the area of transportation of radioactive materials, DOE will furnish a certified transportation package for TVA's use in preparing irradiated TPBARs for transportation. DOE will be the shipper of record. TVA's scope includes preparing the irradiated TPBARs for transportation by loading irradiated TPBAR consolidation containers into a certified transportation package, loading the package onto the transport vehicle, and preparing shipping papers for DOE. TVA will implement the applicable portions of TVA's NRC-approved Radioactive Material Package Quality Assurance Plan associated with use of licensed/certified transportation packages, including that the package supplier is a TVA accepted supplier.

1.5.3 Compliance with DNB Criterion

Action

NUREG-1672, Section 2.4.4, "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."*

Response

During its review of the TPCTR, the NRC staff identified compliance with the DNB criterion as an interface issue for which plant-specific information would be required in the licensee's submittal to support an amendment to the facility operating license for authorization to operate a tritium production core. The acceptability of the limiting core power distributions with respect to DNB performance was explicitly evaluated for the SQN 96-feed maximum TPBAR first transition and equilibrium fuel cycles. The evaluation was performed using the standard approved reload analytical methods described in Reference 1.5.3.1 and is described in more detail in section 2.4.3. The results of the evaluation show that the presence of the TPBARs can be accommodated at the power uprate condition of 3455 MW_t without violating the DNB design bases. The presence of TPBARs in the reload core design did not challenge the DNB criterion. An explicit check of the DNB criterion is included in the cycle-specific reload safety evaluation performed for each SQN reload core. Continued performance of this check will validate the acceptability of each reload core for operation within the DNB design limits.

References

1.5.3.1 Core Operating Limit Methodology for Westinghouse PWRs, BAW-10163P-A, B&W Fuel Company, Lynchburg, Virginia, June 1989.

1.5.4 Reactor Vessel Integrity Analysis

Action

NUREG-1672, Section 2.5.3, "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 plants 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."

Response

Several analyses are performed to determine the impact that neutron irradiation has on the SQN Unit 1 and 2 Reactor Vessel (RV) integrity. These analyses include a surveillance capsule withdrawal schedule, heatup and cooldown pressure-temperature limit curves, pressurized thermal shock calculations and upper shelf energy evaluations. All of these analyses and evaluations can be affected by changes in the neutron fluences and operating temperatures and pressures. The evaluation of the tritium production core assumes that the 1.3% power uprate program has been implemented, and therefore, the impact of the tritium production core is compared to the results of the 1.3% power uprate.

The most critical area is the beltline region of the RV since it is predicted to be most susceptible to neutron damage. The beltline region is defined in ASTM E185-82 (Reference 1.5.4.1) as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material".

Input Parameters and Assumptions

Inlet Temperature

The basis of the equations and tables from Regulatory Guide 1.99, Revision 2 (Reference 1.5.4.2) and 10 CFR 50.61 (Reference 1.5.4.3), which are used in the RV integrity analyses, comes from ASTM E900 (Reference 1.5.4.4). Paragraph 1.1.4 of ASTM E900 stipulates that these equations are valid only in the temperature range of 530 to 590°F. Therefore, the inlet temperature (T_{COLD}) must be maintained within this range to uphold the existing analyses. T_{COLD} for the SQNTPC is 544.8°F (see Table 1-1), which is within the range of validity. Thus, the equations used in the analyses remain valid.

Fluence Projections

Calculated and best estimate fluence values were determined for SQN Units 1 and 2 reactor vessels. These were projected to operating times of 20, 32, and 48 EFPY, assuming cycles starting with cycle 11 are run with a tritium production core and at a reactor power uprated to 3455 MW_t. Calculated fluence values were determined from 2-dimensional neutron transport calculations by a 3-dimensional synthesis technique as recommended in NRC Regulatory Guide 1.190. The best estimate fluence values were determined using a bias factor calculated by comparing calculated surveillance capsule exposure values to a least squares evaluation of measured surveillance capsule dosimetry.

Based on this analysis, it was determined that the maximum vessel exposure point has a lower fluence with the tritium production core fluence projections than for the previous projections made for the 1.3% Power Uprate program.

In a typical low leakage loading pattern, the assemblies on the periphery are mostly low reactivity, twice-burned assemblies that naturally operate at very low powers. This kind of loading pattern limits the accumulation of fluence on the reactor vessel. Because of the larger feed batch (up to 96 assemblies) used in the example equilibrium cycle SQNTPC, the burned assemblies placed on the core periphery are only once-burned and therefore more reactive. To mitigate the potential impact this would have on the vessel fluences and consequently vessel lifetime, the SQNTPC designs that have been developed use one or both of the following methods to reduce the power production in peripheral core locations:

1. Fuel assemblies with higher burnups are loaded into key peripheral core locations,
2. Burnable Poison Rod Assemblies (BPRAs) containing 3.5 w/o B_4C in Al_2O_3 (typical) are loaded in eight peripheral core locations for vessel fluence control.

For the first transition cycle, only the first measure is needed because the fuel burnup is sufficiently high in twice-burned fuel assemblies that BPRAs are not required to meet the criterion. For subsequent transition cycles and the equilibrium cycle both methods are employed due to the lower burnup of once-burned fuel assemblies available for placement in core locations B13 and C14, as well as the symmetric core locations. The locations of the BPRAs in the transition and equilibrium core are shown in Figure 1.5.4-1. The actual tritium production core implementation may involve a lower number of feed assemblies; however, the cycle specific core designs will employ power suppression techniques which may include method 1 and/or 2 to suppress the power in critical peripheral assemblies as required.

Applicable Analyses

Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions. The fluence projections for the SQNTPC do not exceed the fluence projections for the 1.3% uprated power for SQN Units 1 and 2. Therefore, the withdrawal schedules applicable to the uprated core designs without TPBARs remain valid for the tritium production core designs.

Heat-up and Cooldown Pressure - Temperature Limit Curves

A review of the applicability dates of the heatup and cooldown curves for the pressure and temperature limits was performed. This review was accomplished by comparing the fluence projections used in the calculation of the Adjusted Reference Temperature (ART) for all the beltline materials in the reactor vessel for the uprated power conditions to the fluence based on the tritium production design conditions.

Since the revised fluence projections do not exceed the fluence projections used in developing the ART values for the uprated power conditions, the applicability dates for the heatup and cooldown curves for the uprated power conditions remain valid for the tritium production core design.

Pressurized Thermal Shock (PTS)

The RT_{PTS} values for the uprated power conditions do not exceed the screening criteria of the PTS Rule. Since the fluence projections at the tritium production core design conditions do not exceed the fluences used in developing the RT_{PTS} values for the uprated power, the RT_{PTS} values for the tritium production core designs will remain below the NRC screening criteria.

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits (Reference 1.5.4.5) were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at end of life. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The limiting material for SQN Unit 1 is the Lower Shell Forging, while the limiting material at SQN Unit 2 is the Intermediate Shell Forging. SQN Unit 1 is in Category II and SQN Unit 2 is in Category I for the uprated power conditions without TPBARs. Since the fluence projections at the tritium production core design conditions do not exceed the fluence projections for the uprated power conditions without TPBARs, the ERG categories will be unchanged for SQN Units 1 and 2 with tritium production cores .

Upper Shelf Energy (USE)

Based on the 1.3% uprated conditions, all beltline materials in SQN Units 1 and 2 are expected to have an upper shelf energy (USE) greater than 50 ft-lb through end of license (EOL, 32 EFPY), as required by 10 CFR 50, Appendix G (Reference 1.5.4.6). The EOL (32 EFPY) USE values were predicted using the EOL 1/4T fluence projections. Since the fluence projections at the tritium production core design conditions do not exceed the fluence projections for the uprated power conditions without TPBARs, the current predicted USE values for SQN Units 1 and 2 remain valid.

Conclusions

It is concluded that the tritium production core will not have a significant impact on the reactor vessels in SQN Units 1 and 2 based on the following:

1. The core design employs power suppression techniques which may include the insertion of BPRAs in key peripheral fuel assembly locations so that the power in those locations remains comparable to that in the current Sequoyah loading patterns.
2. The inlet temperature for the tritium production core remains within the range of validity for the RV integrity analysis equations.
3. The fluence projections for the tritium production core are bounded by the existing fluence projections for SQN. Therefore, the existing RV integrity analyses remain valid for the Tritium Program.

References

1.5.4.1 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", E706 (IF), in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1993.

1.5.4.2 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.

1.5.4.3 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

1.5.4.4 ASTM E900, "Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)", Reapproved 1994.

1.5.4.5 Emergency Response Guidelines – Revision 1B, Westinghouse Owners Group, February 28, 1992.

1.5.4.6 10 CFR 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 29, 1995.

1.5.5 Control Room Habitability Systems

Action

NUREG-1672, Section 2.6.1, "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."

Response

The acceptance criteria for habitability of the Main Control Room following a design basis accident are based on meeting the relevant requirements of General Design Criteria (GDC) 4, 5, and 19 of 10 CFR Part 50 Appendix A. The documented design basis for the Sequoyah Nuclear Plant Main Control Room systems provides adequate protection of Control Room personnel for operation with a conventional (non-tritium producing) core. The NRC in the SER written for the DOE Topical Report on the reference plant concurred that only the radiation dose criteria are potentially affected by the incorporation of the TPBARs. The NRC noted that the major habitability concern for the referenced plant was the direct consequence of the assumed high leak rate from the Emergency Core Cooling System (ECCS). The 2 gpm assumed leak rate is the value formerly used as a default for plants without a leakage reduction system. The ECCS leakage normally assumed in accident assessments is twice the leak rate that triggers corrective action under the applicable leak reduction program. The NRC further noted that values of 2 gallons per hour or less which are typically used would meet the relevant dose criterion.

Analysis to be provided later

References

1.5.5.1 Federal Guidance Report No. 11, LIMITING VALUES OF RADIONUCLIDE INTAKE AND AIR CONCENTRATION AND DOSE CONVERSION FACTORS FOR INHALATION, SUBMERSION, AND INGESTION. EPA-520/1-88-020. U.S. EPA. Washington, DC 1988.

1.5.5.2 Federal Guidance Report No. 12, EXTERNAL EXPOSURE TO RADIONUCLIDES IN AIR, WATER, AND SOIL. EPA 402-R-93-081 U.S. EPA. Washington, DC 1993.

1.5.6 Specific Assessment of Hydrogen Source and Timing of Recombiner Operation

Action

NUREG-1672, Section 2.9.2, "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."

Response

Introduction

The acceptance criteria for the design of the systems provided for combustible gas control are the relevant requirements of 10 CFR Part 50, Paragraphs 50.44 and 50.46 and General Design Criteria 5, 41, 42, and 43. As part of these acceptance criteria, analyses should indicate that a single system train is capable of maintaining the combustible gas concentrations to levels such that uncontrolled hydrogen/oxygen recombination would not take place.

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

- metal-water reaction with the fuel cladding
- corrosion of materials in contact with spray/sump solutions
- radiolysis in the sump and core solutions
- RCS inventory prior to the accident

When operating with a TPC, there are additional sources of post-LOCA hydrogen production that should be considered. They are:

- metal-water reactions with the zirconium components associated with the TPBARs, and
- tritium and hydrogen that exist in the TPBARs prior to the accident.

Although radiolysis, which is a function of decay energy of the fission products, could be marginally impacted by the TPC, the impact is considered to be negligible. This is particularly true since the fuel burnups for a TPC are not significantly different than those associated with conventional cores operating with 18-month fuel cycles.

TPBAR Metal-Water Reaction

One of the potential sources of hydrogen unique to a TPC design is that associated with zirconium getter materials contained within the TPBARs. The zirconium that is subject to the zirconium-water reaction is specified in 10 CFR 50.44 (Reference 1.5.6.1) to be only that associated with the "... fuel cladding surrounding the active fuel region ..." and "... the mass of metal in the cladding cylinders surrounding the fuel ..." (Note: the Sequoyah evaluation conservatively assumes the grid spacers are also subject to the reaction). This follows since it is generally only the metal in the active core region that is subject to the high temperatures (in excess of 1800 °F), which are necessary for the zirconium-water reaction to occur. However, if the TPBAR cladding is breached following a LBLOCA, the potential for a metal water reaction with internal zirconium components can be postulated.

Based on the chemical stoichiometry of the zirconium-water reaction, one pound-mole of zirconium metal reacted must produce two pound-moles of hydrogen. That is, 7.9 standard cubic feet (scf) of hydrogen gas is produced for each pound of zirconium metal reacted. The maximum amount of zirconium associated with the getter material (300 grams per TPBAR) in 2,256 TPBARs (i.e., the total number of TPBARs in an equilibrium cycle in Sequoyah Unit 1 or Unit 2) is 1,492 pounds.

The worst case scenario is to assume that all TPBARs burst and, following expulsion of the gases, some diffusion of steam into the TPBAR could be postulated. For conservatism, the TPBAR internal zirconium components are treated in an analogous fashion to the treatment of the internal surface of fuel rod cladding following clad burst. For a fuel rod, zirconium oxidation is calculated on the internal surface over the length of a three-inch long burst node. For each TPBAR, complete oxidation of the zirconium within a twelve-inch long burst node following a LBLOCA is considered, with the resulting hydrogen released to the containment atmosphere. The fraction of the total absorber length represented by the TPBAR burst node length is

$$F = 12 \text{ in} / 126 \text{ in} = 0.0952$$

where a TPBAR absorber length of 126 inches is used in order to conservatively estimate the fraction. The value determined above is equal to the fraction of the total TPBAR zirconium mass involved in the reaction. Then, the equivalent hydrogen that could be released is

$$V' = 1,492 \times 0.0952 \times 7.9 = 1,122 \text{ scf}$$

TPBAR Tritium and Hydrogen Inventories

Another potential contributor to the hydrogen inventory associated with a TPC is the hydrogen (including tritium) inventory contained within the TPBARs that would be available for release. For conservatism, it is assumed that the maximum tritium gas inventory is released to containment.

Conservatively assuming the design limit of 1.2 grams per rod at the end of the fuel cycle, the equivalent volume of tritium gas (T_2) associated with the mass of tritium contained within the 2,256 TPBARs in the core is 357 ft3 of T_2.

An additional source of hydrogen associated with the TPBARs is that generated from the $^3He(n,p)T$ reaction inside the rods. At end of a fuel cycle, this source could generate an additional 16 scf, which would also be available for release following a LBLOCA.

Results and Conclusions

The additional hydrogen inventories that are conservatively estimated to be associated with a TPC are 1,122 scf associated with zirconium-water reactions with the TPBAR getter materials, 357 scf of tritium gas from the TPBARs, and 16 scf of hydrogen from $^3He(n,p)T$ reactions inside the rods. This sums to a total of 1,495 scf as the potential additional amount of hydrogen contributed by the TPBARs following a LBLOCA.

This inventory would be expected to exist in the primary coolant as water or tritiated water (HTO or T_2O), rather than as a gas. However, even if the complete hydrogen/tritium inventory associated with a TPC is conservatively assumed to be released to the containment atmosphere as gas, the added inventory represents only a 4% increase in the amount of hydrogen gas in the containment one day after a LBLOCA. That is, the total inventory in the containment at one day after a LBLOCA, including TPC sources is 36,898 scf, which is 4% higher than the value of 35,403 calculated on the basis of operation with a conventional core.

The lower flammability limit for hydrogen in the containment atmosphere that should not be exceeded as defined in USNRC Regulatory Guide 1.7 (Ref. 1.5.6.2) is 4 volume percent. For a Sequoyah plant with a total containment free volume of 1,230,000 ft3 a concentration of 4 volume percent equates to approximately 49,200 scf of hydrogen. Thus, the contribution of the TPC tritium inventory to the amount of hydrogen associated with the recommended Regulatory Guide limit is only about 3%, i.e.,

$$F' = 1{,}495 / 49{,}200 = 0.030$$

It is concluded that even based on highly conservative assumptions, the TPBARs are not a significant contributor to the post-LOCA hydrogen inventory. The TPC will not have a significant impact on the total hydrogen production and concentrations within the containment, as compared to the values associated with operation with a conventional core. The maximum hydrogen concentration with a TPC can be maintained at less than the lower flammability limit of 4 volume percent, with one recombination train in operation.

References

1.5.6.1 USNRC Code of Federal Regulations, 10CFR Part 44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors".

1.5.6.2 USNRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident", Revision 2, November 1978.

1.5.7 Light – Load Handling System

Action

NUREG-1672, Section 2.9.1, "DOE evaluated the effect 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."

Response

The TPBAR consolidation and shipping phase of the program was considered to be beyond the scope of the TPCTR (Section 2.9.2). However, it has been evaluated with respect to the light load handling system. The handling of items during TPBAR consolidation will be performed by using the Spent Fuel Pit Bridge crane, which utilizes a specialized fixture and tooling to transport the TPBAR assemblies, consolidate individual rods into consolidation canisters, dispose of empty baseplates, transport the canisters for storage in the Spent Fuel Pit, and finally load canisters into shipping casks for transport off-site.

The weight of a fuel assembly with 24 TPBARs and its hold-down plate (63 additional lbs for TPBARs) is less than a fuel assembly with a Rod Control Cluster Assembly (RCCA) and therefore is bounded by the current assumed weight of assembly for purposes of analyzing fuel handling and storage facilities. The fuel assembly with TPBARs has the same external configuration as a fuel assembly without TPBARs allowing for interface with existing fuel handling/storage equipment. Additionally, this weight is conservative for purposes of defining a NUREG-0612 "Heavy Load".

During consolidation of TPBARs from a baseplate, rods are released from the baseplate one at a time. (For a description of the consolidation process see Section 1.5.1). Additionally, the consolidation fixture is designed to seismic category 1(L) to preclude damage to consolidated TPBARs while in the fixture and to the spent fuel pool liner. After approximately 300 rods are released into a canister, the loaded canister is transported to a designated spent fuel pool cell location using a canister handling tool suspended from the SFP Bridge crane. Since damage to more than 24 TPBARs has not been evaluated, handling of the loaded canister with the following analysis/design features will limit, to an acceptable level, the possibility of damage to more than 24 TPBARs during handling:

1. In accordance with NUREG-0612, -0554 and ANSI N14.6, the Spent Fuel Pit Bridge crane and canister lifting device will contain sufficient aspects of the single failure proof criteria to preclude a drop of the loaded canister as delineated below.
    a) The SFP Bridge crane is considered equivalent-single-failure proof with respect to structural integrity in accordance with NUREG-0612 (NUREG-0554) due to the following:
        1) Since the SFP Bridge crane has a capacity of 2000 lbs. and the weight of the submerged loaded canister is approximately 700 lbs., the crane has safety factors twice the normally required values.
        2) The crane is equipped with redundant high hook limit switches of different designs to preclude structural failure.
    b) The lifting tool is provided with a safety lanyard to limit canister descent in the fuel pool to such an extent that spilling of the TPBARs out of the open topped canister, if the canister bottom were to hit an obstruction and cause the canister to tip, is prevented. The lanyard is sized to stop the canister from a maximum hook speed of 40 fpm. Administrative requirements require that the safety lanyard be attached to the lifting tool during hoisting when the canister is not engaged in a SFP rack cell, the consolidation fixture holster, or cask by at least 12".
    c) In accordance with ANSI N14.6 sections for Critical Loads, the lifting tool is designed to twice the normal safety factors, tested to twice the normally required loads, and inspected utilizing required NDE methods, thereby the tool is considered equivalent-single-failure proof. It will also have an air actuated fail-closed safety latch to prevent the tool hook from disengaging from the canister lifting bail.

2. The loaded canister weight and its handling tool is less than that of a fuel assembly and its handling tool. Additionally, due to the design features listed above, the canister descent is limited to an uncontrolled lowering (e.g. a control failure) of a canister at a maximum hoist speed of 40 feet per minute, thereby limiting the kinetic energy to less than that of the fuel assembly. Therefore, fuel assembly drop accidents in the pool remain bounding .

3. An analysis has been performed to demonstrate that damage to more than 24 TPBARs contained in a canister is precluded for all credible impact scenarios during canister handling.

4. The drop of the light-weight, base-plate with TPBARs, within the spent fuel pool/cask load pit area, is bounded by the analysis of a fuel handling accident damaging an irradiated fuel assembly and 24 included TPBARs.

1.5.8 Station Service Water System

Action

NUREG-1672, Section 2.9.1, "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."

Response

Introduction

The design basis function of the Station Service Water System, which is called the Essential Raw Water Cooling System (ERCW) for SQN, includes providing a cooling loop for heat removal from the Component Cooling System (CCS). The ERCW supplies water from the Ultimate Heat Sink (UHS) (Tennessee River) to cool primarily safety related components. The CCS is the primary means for cooling the plant and removing residual decay heat during late stages of plant cooldown and during outages. The CCS intermediate cooling loop provides a heat sink to the Spent Fuel Pool Cooling and Cleanup System (SFPCCS) and Residual Heat Removal (RHR) system.

Tritium Impact on Spent Fuel Pool Decay Heat

TVA has prepared a quantitative analysis of expected spent fuel decay heat for both Tritium Production Core (TPC) and non-TPC cores. The analysis is based on comparative decay heat data prepared by TVA for a base non-tritium core, a TPC with 80 fresh fuel assembles (80-feed), and a TPC with 96 fresh fuel assemblies (96-feed). The results of the analysis show that the 80 feed case was limiting for decay heat (i.e, freshly offloaded core), and the 80-feed TPC core contributes a slightly higher decay heat over the non-TPC and the 96-feed TPC, due to isotopic composition differences between the base and TPC cores, for the same design basis reactor power level. The results of the analysis show that the 96-feed case was limiting for residual SFP heat (i.e., heat coming from total of previously discharged assemblies). TVA has assumed the worst case combination of these two heat sources. The TVA analysis has quantified the actual TPC impact on core heat loads at approximately 0.5 MWt, which included both the decay heat generated by freshly discharged fuel assemblies during a refueling outage, and the additional residual decay heat from the increased discharge rate (96 per outage) of fuel assemblies into the pool. This value is based on conservative, full pool SFP conditions.

Increased Spent Fuel Pool Cooling Heat Rejection on ERCW

The design basis analysis for the ERCW was evaluated for impact from the increased heat load from the CCS. The increased SFPCCS heat load rejection to the CCS will not result in a significant temperature increase in ERCW. The higher proposed increase in allowable decay heat load in the SFP is comprised of both TPC related decay heat increase and additional margin to allow off loading fuel to the SFP as early as 100 hours. The increase in decay heat associated with TPC is approximately 1.7 MBTU/Hr. The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 8 MBTU/Hr. The proposed increase in decay heat above the approximate 1.7 MBTU/Hr associated with TPC, is decay heat that is shifted from the RHRS to the SFPCCS. The shifting results from the fact that fuel is either in the core being cooled by RHRS, or it is in the SFP being cooled by the SFPCCS. Since the decay heat has only shifted between systems, there is no net increase in CCS heat load on the ERCW system for this portion of the increased decay heat.

The design basis thermal analysis of record for the ERCW has sufficient margin to accommodate the increased CCS heat loads resulting from increased SFPCCS allowable decay heat loads. The increase in decay heat load is well within the design bases limiting heat load imposed on the ERCW during other modes of operation. Increased ERCW flows are the same higher flow rates that have been specified during other modes of operation. This small amount of increased decay heat and increased ERCW flow, when compared to the overall flow rates through the ERCW System, produces an insignificant increase in ERCW temperature (< 0.1°F) leaving the plant site.

The additional heat load rejected to the ERCW from the CCS heat exchanger results in minimally elevated piping temperatures. The downstream dilution effect, however, minimizes the impact of the elevated ERCW temperatures, as nearly all ERCW flows return to one of two headers prior to being discharged from the plant. The increased thermal loading on the piping analysis and support analysis of the ERCW System is well within existing design temperatures.

Conclusions

The ERCW System has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. The ERCW system can also accommodate the additional SFP heat loads imposed by the proposed change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities will not have an adverse impact on the ERCW heat removal capabilities. For additional information on the SFPCCS, see Section 1.5.11.

1.5.9 Ultimate Heat Sink

Action

NUREG-1672, Section 2.9.1, "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."

Response

Introduction

The design basis function of the UHS is to provide an uninterrupted source of cooling water for decay heat removal. The maximum allowable inlet temperature for the UHS is 84.5°F. The ERCW System is utilized to supply water from the UHS to cool primarily safety related components. The CCS is the primary means for cooling the plant and removing residual decay heat during late stages of plant cooldown and during outages via its intermediate cooling loop providing a heat sink to the SFPCCS and RHR system.

Tritium Impact on Spent Fuel Pool Decay Heat

See previous discussion under Interface Item 1.5.8.

Increased Spent Fuel Pool Cooling Heat Rejection on UHS

The design basis analysis for the UHS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant temperature increase in the

UHS. The increase in decay heat associated with TPC is approximately 1.7 MBTU/Hr. The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 8 MBTU/Hr. This total increase in decay heat load is well within the design bases limiting heat load imposed on the ERCW and UHS during other modes of operation. Increased ERCW flows are the same higher flow rates that have been specified during other modes of operation. This small amount of increased decay heat and increased ERCW flow, when compared to the overall flow rates of the UHS through the ERCW System, produces an insignificant increase (< 0.1°F) in UHS temperature leaving the plant site. Since there is no significant increase, and since the ERCW has significant margin available, no changes to the ERCW temperature requirements are warranted.

Conclusions

The UHS has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. The UHS can also accommodate the additional SFP heat loads imposed by the proposed change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities at SQN will not have an adverse impact on the UHS heat removal capabilities. For additional information on the SFPCCS see Section 1.5.11.

1.5.10 New and Spent Fuel Storage

<u>Action</u>

NUREG-1672, Section 2.9.2, "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."

Response

This item will be addressed in a separate submittal by TVA.

1.5.11 Spent Fuel Pool Cooling and Cleanup System

Action

NUREG-1672, Section 2.9.3, "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 SFPCCS. 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."

Response

Introduction

The SFPCCS for SQN is sized to handle full core off-loads. In the 1994-95 timeframe, SQN underwent spent fuel storage rack additions, which included development of a new thermal hydraulic analysis based on standard NRC approved methodologies which are scenario based. After the rerack design change TVA recognized the impracticality of following a scenario based set of limits during plant operation for predicting SFP decay heat load. Following the licensing efforts associated with the rerack modification at SQN, the FSAR was revised to capture a limiting value of decay heat that could be placed in the SFP, based on outage specific decay heat analysis performed for each outage. This approach provided a more realistic means (based on quantitative limits instead of scenario based limits) of assuring compliance with the maximum allowable design basis decay heat loads that could be placed in the SFP at any time. Compliance with these limiting values provides assurance that, should a train of SFPCCS fail, maximum analyzed temperatures of the SFP and attendant decay heat removal system piping will not be exceeded.

UFSAR Section 9.1.3 now allows outage specific decay heat values to be used to determine the acceptable point in time that core off loading activities may commence without exceeding the design basis maximum allowable heat load. Prior to each outage, a core specific and real time SFP decay heat assessment is prepared, which considers core operating parameters such as average fuel burn-up, interim trips, and coast-downs, etc. to develop pre-outage data for expected core and SFP decay heat. Procedures are in place to assure that at no time during core off-loading activities will the design basis limits of the SFPCCS be exceeded. Adherence to the established limiting values of allowable SFPCCS decay heat ensures that the maximum SFP temperature does not exceed the pre-established maximum allowable design temperatures.

Tritium Impact on SFP Decay Heat

See previous discussion under Interface Item 1.5.8.

In addition, the impact of the higher heat load in the SFP could be mitigated by delaying the start of core off-load by approximately 15 hours. Therefore from a design basis standpoint, it could be concluded that tritium production operations have no adverse impact on SFP heat loads or the ability of associated systems to remove the heat loads. However, since delaying the start of off-loading of the core during a plant outage results in a financial impact to plant operations, TVA has developed an alternate decay heat analysis which would compensate for this additional heat load and also accommodate core off-loading as early as 100 hours after shutdown.

Alternate SFP Decay Heat Analysis

An alternate analysis has been prepared by TVA to predict SFP transient thermal performance. This alternate analysis represents a change in methodology from the current analysis. The alternate analysis utilizes the same basic methodology, equations, and /or data as the current analysis, which was prepared in support of the previously licensed rerack effort. The alternate analysis, however, utilizes a modified methodology which allows varying SFP heat exchanger fouling and varying SFP heat exchanger coolant (CCS) temperature, to perform thermal balances on the SFP. Heat added by both core decay heat and residual decay heat from previously discharged batches provide the heat input parameter for the analysis. Since the new analysis is primarily an overall system heat balance, the source or mechanism for predicting actual core decay heat becomes less important. The new analysis models core decay heat post shutdown utilizing conservative core burnup generated using Nuclear Fuels computer code DHEAT, which is based on ANSI/ANS-5.1-1994, REG GUIDE 3.54, and NUREG/CR-2397. The overall system heat balance models SFP heat removal by the same two mechanisms as utilized in the existing analysis of record, via SFP heat exchangers and evaporative losses to ambient.

SFP Heat Exchanger Fouling Factor

The analysis of record utilized design fouling factors of 0.000575 for the tube and 0.0005 for the shell side fouling. Actual fouling of the SFP heat exchangers has been found to be considerably less than design, with minimal negative trending over a long period of time, based on Sequoyah experience. This experience is consistent with expectations, given that both the CCS and the SFPCCS streams are clean water systems, approaching demineralized water in purity and clarity. The conditions required for fouling of the heat exchanger are not present in this application. Actual data to date from SQN suggest low fouling rates of the heat exchanger over 20 years without cleaning. The use of this new methodology will require the use of certified Measuring and Test Equipment (M&TE) under written procedures for the determination of heat exchanger fouling factors prior to taking credit for lower fouling. Sufficient testing will be performed to clearly establish the presence of any fouling trend. Due to the high purity of the coolant and cooled streams, and the proven history to date of low fouling, high fouling rates or other deviations to any established trend are not likely. Analysis performed with less than design fouling indicated significant benefit can be obtained in removing additional heat load from the SFP.

Component Cooling System Maximum Water Temperature

The analysis of record utilized design maximum values for CCS temperatures for the cooling medium on the shell side of the SFP heat exchangers. The maximum design temperature for CCS during refueling outages is 95°F. This value, however, is very conservative relative to the actual amount of heat being rejected to the CCS. The design basis for the CCS included significantly higher decay heat loads based on Residual Heat Removal (RHR) system heat loads shortly after shutdown. By the time the core is completely off-loaded (approximately 136 hours after shutdown), the RHR heat load is essentially zero. By increasing the flow of ERCW to the CCS heat exchanger to its maximum allowable flow, CCS maximum temperature can be decreased to values less than the 95°F design value, based on design ERCW temperature and design fouling of the CCS heat exchanger.

Results of Alternate Analysis

By performing several analyses of SFP thermal performance at varying fouling factors from 0.0005 to 0.0001 and decreased CCS temperatures, a series of curves have been developed to provide operator guidance for an increase in allowable SFP decay heat. An analysis was performed for the limiting case of single train operation, in which the allowable design heat load was increased up to a maximum without exceeding the maximum design SFP temperature. Final curves of allowable decay heat vs. CCS Temperature and SFP Heat exchanger fouling were developed which included margin to account for inaccuracy inherent in reading graphs, and to add additional modeling conservatism. To implement these changes, SQN's design change process requires procedures to be developed or existing procedures reviewed and revised, if necessary, to allow increased decay heat to be placed in the SFP based on actual values for CCS temperature and SFP heat exchanger fouling. The following is a tabulation of specific SFP design values and parameters for both the existing design and the proposed alternate design.

SQN SPENT FUEL POOL DESIGN PARAMETERS		
Parameter	**Existing Design Value**	**Proposed Value (Alternate Analysis)**
Maximum Allowable Decay Heat Load	45.37 MBTU/Hr	45.37 - 55 MBTU/Hr See Note 1.
SFPCCS Flow	2300 GPM per Hx	2300 GPM per Hx
CCS Flow	3000 GPM per Hx	3000 GPM per Hx
Allowable Tube Plugging	5 %	5 %
Tube-Side Fouling (hr*ft^2*°F/Btu)	0.000575	0.0005 - 0.0001
Shell-Side Fouling (hr*ft^2*°F/Btu)	0.0005	0.0005 - 0.0001
Maximum CCS Temperature	95°F	95 - 80°F (Note 1)
Maximum SFP Temperature (2-Train)	144°F	144°F
Maximum SFP Temperature (1-Train)	183°F	183°F
Minimum Time to SFP Boiling	2.64 Hours	1.14 Hours
Average SFP Heat-Up rate	10.98°F/Hr	25.35°F/Hr
Maximum Boil-Off Rate	103 GPM	118.2 GPM
Time until only 10 feet of water over racks - without makeup	30 Hours	25.7 Hours
Time until only 10 feet of water over racks - with 103 gpm makeup	See Note 2	See Note 2
Margin to Localized Rack Boiling	4.80°F	3.5°F
Departure from Nucleate Boiling at maximum heat load and maximum SFP temperature.	No	No
Notes:		
1. The range of values represent allowable heat loads based on specific combinations of heat exchanger fouling between 0.0005 and 0.0001 (hr*ft^2*°F/Btu) and actual CCS temperatures between 95 to 80°F.		
2. Analysis has shown that SQN has a qualified source of makeup water of 103 GPM, therefore the 10 feet above rack level is never reached for the Boil-Off rates determined.		

Impact of Higher Allowable Decay Heat in the SFP

As shown in the table above, the proposed change will not result in an increase in maximum SFP temperature. The only operational effect is noted during complete loss of both trains of cooling, whereby the higher allowable decay heat results in higher boil-off rates and faster required response times to mitigate the loss of SFP cooling event. The proposed values above, however, are reasonable and ample time exists to take appropriate action to introduce makeup water to the SFP from one of multiple sources.

An analysis has also been performed to evaluate the affect on localized temperatures within a spent fuel rack. The analysis was performed consistent with existing analysis methodologies except the rack and pool area were modeled using a three dimensional nodalization, instead of two dimensional. The inputs were revised to be consistent with the maximum allowable decay heat value (55 MBtu/hr). The results of the analysis show that while the margin to localized boiling has decreased, localized boiling within a rack will not occur. The analysis specifically concluded that:

1. the maximum local water temperature in the fuel storage racks was less than the local saturation temperature of the water, and
2. The maximum fuel clad temperature, while greater than the local water saturation temperature, would not result in departure from nucleate boiling (DNB), and that fuel cladding integrity would be maintained.

The increased heat load on CCS during single or dual train operation has minimal impact and is well within the design limits of the CCS system. Conservatism is maintained in the alternate analysis by ignoring all heat losses through concrete walls and SFPCCS piping, and ignoring both the mass of metal racks and fuel in the SFP and the mass of water in the transfer canal when determining the SFP heat capacity. The proposed change will not result in exceeding any system design limitation.

While existing design limits & operational procedures are adequate to prevent exceeding design limits on allowable SFP heat load, TVA proposes to revise the allowable heat loads. TVA proposes to increase the maximum allowable decay heat in the SQN SFP from 45.37 MBTU/Hr to a range between 45.37 MBTU/Hr and 55 MBTU/Hr. The lower value of 45.37 MBTU/Hr will only be exceeded if actual operating conditions of lower CCS temperature and/or lower than design fouling is present. Specific curves relating CCS Temperature and SFP heat exchanger fouling to allowable SFP decay heat have been developed to assist Operations in evaluating allowable SFP decay heat for each core off-loading evolution. These higher values of allowable decay heat within the SFP will not result in exceeding the analyzed maximum SFP temperature under normal full core off-load conditions (two train operation) of 144°F, and a faulted maximum temperature (one train operation) of 183°F. As described in Enclosure 2, TVA is seeking a licensing change to its SFPCCS allowable heat loads to allow use of actual fouling factors and CCS temperature in lieu of design values.

Conclusions

The SFPCCS has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. Without this change in methodology, existing SFPCCS operational parameters can accommodate Tritium Production operations by delaying the start of off-loading the core until design allowable heat loads can accommodate core and residual decay heat. The SFPCCS can also accommodate the additional SFP heat loads imposed by the proposed change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities will not have an adverse impact on the SFPCCS heat removal capabilities.

1.5.12 Component Cooling Water System

Action

NUREG-1672, Section 2.9.4, "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."

Response

Introduction

TPCTR Section 2.9.4 addressed impacts on the Component Cooling System (CCS). The report concluded that the actual impact to CCS heat removal capacity was primarily influenced by the increase in SFPCCS decay heat. The report suggested that the extent of the Spent Fuel Pool Cooling and Cleanup System (SFPCCS) impact on the CCS system would depend on available margins in the system design, if any, and should therefore be evaluated on a plant-specific basis.

SER Section 2.9.4 indicated that the primary concern of the TPC impact on CCS was the additional heat load imposed by the SFPCCS on CCS, and any required changes to flow to meet the increased heat removal demand. The SER also indicated that if the impact on CCS was significant, the ability of the CCS to serve other safety related heat exchangers (e.g. Residual Heat Removal System (RHRS)) may be affected.

The design basis functions of the CCS include providing an intermediate cooling loop for heat removal from several safety related radioactive system heat exchangers, as well as several non-safety related components. Two of the highest heat loads placed on the CCS include the SFPCCS and the RHRS. These two decay heat systems are the primary means for cooling the plant and removing residual decay heat during later stages of plant cooldown and during outages.

Tritium Impact on Spent Fuel Pool Decay Heat

TVA has prepared a quantitative analysis of expected spent fuel decay heat for both TPC and non-TPC cores. The analysis is based on comparative decay heat data prepared by TVA for a base core, an 80-Feed TPC, and a 96-Feed TPC. The results of the analysis show that the 80 feed case was limiting, and the 80-Feed TPC core contributes a slightly higher decay heat over the non-TPC and the 96-Feed TPC, due to isotopic composition differences between the base and TPC cores, for the same design basis reactor power level. The TVA analysis has quantified the actual TPC impact on core heat loads at approximately 1.7 MBTU/Hr, which included both the decay heat generated by freshly discharged fuel assemblies during a refueling outage, and the additional residual decay heat from the increased discharge rate (96 per outage) of fuel assemblies into the pool. This value is based on a conservative, end of life SFP conditions.

Increased Spent Fuel Pool Cooling Heat Rejection on CCS

The design basis analysis for the CCS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant temperature increase on CCS. The increase in decay heat associated with TPC is approximately 1.7 MBTU/Hr. This decay heat load increase is less than 2% of the total design heat load on the CCS. The higher proposed increase in allowable decay heat load in the SFP, however, is comprised of both TPC related decay heat increase, plus additional margin to allow commencement of core off loading activities as early as 100 hours after shutdown. The proposed increase in decay heat above the approximate 1.7 MBTU/Hr associated with TPC, is a CCS heat load that is shifted from the RHRS to the SFPCCS. The shifting results from the fact that fuel is either in the core being cooled by RHRS, or it is in the SFP being cooled by the SFPCCS, both systems ultimately rejecting their respective heat burdens on the CCS.

CCS design thermal analysis have been revised to reflect increased SFPCCS allowable decay heat loads. CCS flows to the SFPCCS heat exchangers have not been increased. The additional heat load rejected to the CCS from the SFPCCS heat exchanger results in slightly elevated CCS temperatures, but are well within existing design basis values. Piping analysis and support analysis of the CCS have been previously analyzed at a higher ultimate temperature associated with more bounding operational modes, and are not affected by the increased CCS heat load. The mixing of multiple CCS return lines into common headers minimizes the impact of the elevated CCS temperatures, since as SFPCCS heat loads increase, the RHRS heat loads decrease. With all CCS flows returning to a common header prior to returning to the CCS/ERCW heat exchangers, there is no measurable change to the mixed stream CCS temperature.

Impact on ERCW due to Increased Spent Fuel Pool Cooling Heat Rejection on CCS

Since higher allowable SFP decay heat can be placed in the SFP if CCS temperatures and /or SFP heat exchanger fouling factors are shown to be less than design, maintaining the CCS temperature during outages to as low as possible is desired. CCS temperatures can be lowered considerably if ERCW flows to the CCS heat exchangers are increased. Plant operations will be provided operating guidance to assist with ERCW flow requirements to the CCS heat exchangers to keep CCS temperatures as low as possible during periods of fuel off-load. The increased ERCW flow rates are within existing flow criteria established for other modes of operations.

Conclusions

The Component Cooling System has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. The CCS can also accommodate the additional Spent Fuel Pool heat loads imposed by the proposed change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities will not have an adverse impact on the CCS heat removal capabilities. Additional information on SFP decay heat is provided in response to NRC SER Interface Item 11.

1.5.13 Demineralized Water Makeup System

Action

NUREG-1672, Section 2.9.5, "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."

Response

The SER and TPCTR Section 2.9.5 addressed possible impacts on the Demineralized Water Makeup System (DWMS). This section acknowledged that tritium production activities would result in increased tritium levels in the Reactor Coolant System (RCS). To maintain tritium levels within the RCS at current levels, additional feed and bleed operations may be required. Any increase in feed and bleed operations requires additional demineralized water as makeup. The SER required the specific impact on DWMS from increased feed and bleed demand be evaluated.

TVA does not intend changes to the plant's current feed and bleed operations to control boron concentration in the RCS. Continuation of the current feed and bleed program will result in the RCS observed maximum tritium levels of 2.5 μCi/gm increasing to around 9 μCi/gm with the TPC. This increase is due to normal reactor tritium production plus the tritium permeation from TPBARs.

Information to be provided later

In the abnormal event of two TPBAR failures, RCS tritium values could increase to approximately 105 μCi/gm. Following this unlikely event, approximately 150,000 gallons of additional feed and bleed would be necessary to reduce the tritium concentration to the 9 μCi/gm range. This estimate is based on the failures occurring near the end of the cycle.

Information to be provided later

Within the SQN DWMS there exists sufficient surge capacity as well as production capacity to meet these projected needs. As tritium levels increase in the RCS, ample planning time will be available to assure adequate surge volume is available and production rates are capable of meeting demand.

SQN uses vendor supplied equipment to produce high purity water for use in the site DWMS. The capacity at SQN is in the nominal 175 gpm range. Storage of demineralized water exceeds 500,000 gallons in available tanks.

Conclusions

TVA's review of the DWMS for SQN has determined that the current system's storage and water production capacity, compared to the expected increase in feed and bleed required to mitigate a two TPBAR failure event, is adequate.

Information to be provided later

The DWMS and storage tanks will not require modification, nor will the water supply contract require changes to support tritium production activities at SQN. See Section 1.5.14 for more information concerning Liquid Waste Management.

1.5.14 Liquid Waste Management System

<u>Action</u>

NUREG-1672, Section 2.11.2, "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."*

Response

TVA has performed an evaluation and determined that for normal TPBAR operation (permeation only), TVA will maintain normal RCS feed and bleed operation for boron control throughout the cycle. Primary coolant discharge volumes with a TPC will therefore be comparable with current plant practice. The maximum tritium level in the RCS is anticipated to be about 9 μCi/g.

Site-specific data collected during recent extended operating cycles (WBN Unit 1 Cycle 3 and SQN Unit 1 Cycle 10) have provided data from which to estimate the impact of tritium on station radiological conditions. The RCS maximum tritium levels noted during the extended operating cycles were ≈ 2.5 μCi/g with a cycle RCS tritium mean of ≈ 1.0 μCi/g. The TVA experienced end of cycle (pre-flood up) RCS tritium values have typically been in the 0.1 - 0.3 μCi/g range for both WBN and SQN. The post-flood up tritium values have typically been in the mid 10^{-2} μCi/g range. The extended cycle peak RCS tritium values of ≈ 2.5 μCi/g have resulted in containment peak tritium Derived Air Concentration (DAC)-fractions of <0.15 for both WBN and SQN with a containment average DAC-fraction of about 0.08. It is understood that containment tritium DAC values are a function of the RCS tritium activity, the transfer of tritium from the RCS to the containment atmosphere (leak rate), and the turnover/dilution of the containment atmosphere through periodic and continuous containment venting and purging.

The projected tritium release to the RCS with a TPC containing the maximum number of TPBARs (2304) releasing tritium at the design maximum permeation rate will result in about a factor of four increase over the current tritium production rate.

By extrapolation it has been calculated that with no modifications to TVA's current boron-control feed and bleed methodologies, the design basis RCS maximum tritium values will approximate 9 μCi/g with a cycle mean of ≈ 3.6 μCi/g. These values would indicate an estimated containment peak tritium DAC-fraction of ≈ 0.6 and an average containment tritium DAC-fraction of about 0.3. The design basis estimated containment average tritium DAC-fraction equates to an effective dose rate of about 0.7 mrem/h.

The TVA TPC estimated end of cycle (pre-flood up) RCS tritium values are projected to be in the 0.4 - 1.2 μCi/g range.

For TPBAR abnormal operation, TVA will establish two tritium RCS action levels > 9 μCi/g and > 15 μCi/g. The lower action level will require more frequent sampling (once/day) to monitor the RCS tritium levels. In the unlikely event that the higher action level is exceeded, TVA will take further action to minimize the onsite and offsite radiological impacts of abnormal RCS tritium levels. These actions may include but not be limited to: initiating actions to determine cause, more frequent tritium monitoring of RCS as well as other potentially impacted areas such as containment, increased feed and bleed of the

RCS to reduce the tritium concentration, and the temporary onsite storage of tritiated liquids to ensure that the discharge concentration limits are met. The actions levels described above will be used in response to what TVA believes to be extremely unlikely abnormal increases of the tritium levels in the RCS. Plant specific procedures will be developed before TPBAR irradiation utilizing these action levels.

Information to be provided later

Conclusions

TVA's review of normal TPBAR operation (permeation only), has established that TVA will maintain normal RCS feed and bleed operation for boron control throughout the cycle. Primary coolant discharges volumes with a TPC will therefore be comparable with current plant practice. The maximum tritium level in the RCS are anticipated to be about 9 μCi/g.

For TPBAR abnormal operation, TVA will establish two tritium RCS action levels > 9 μCi/g and > 15 μCi/g. The lower action level will require more frequent sampling (once/day) to monitor the RCS tritium levels. In the unlikely event that the higher action level is exceeded, TVA will take further action to minimize the onsite and offsite radiological impacts of abnormal RCS tritium levels.

Information to be provided later

1.5.15 Process and Effluent Radiological Monitoring and Sampling Systems

Action

NUREG-1672, Section 2.11.5, "In Section 2.11.6 of the TPCTR, 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 licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

Response

TVA has reviewed its process and effluent monitoring and sampling equipment program and determined that this program requires minor modifications for a TPC. These changes are limited to the modification of the Auxiliary Building and Shield Building Exhaust tritium sampling from periodic effluent grab samples to continuous effluent sampling during periods of release. Other sample frequency enhancements to the existing monitoring programs are discussed in Sections 2.9.6, 2.11.3 and 2.11.4.

Tritium Monitoring

In this section, the various techniques used to monitor for tritium in gases (primarily air), in liquids are discussed.

Air Sampling

For Tritium air sampling the sampled gas (usually air) must be analyzed for tritium content (usually by liquid scintillation counting). The usual technique is to flow the sampled air through either a solid desiccant (molecular sieve, silica gel, or Drierite) or water or glycol bubblers.

Another available technique for sampling HTO in room air is to use a "cold finger" or dehumidifier unit to freeze or condense the HTO out of the air. When using this methodology, to determine the tritium in air concentration, the relative humidity must be known. A typical lower limit of detection for in-station tritium air samples is 2×10^{-10} µCi/ml.

Liquid Monitoring

Liquids will be monitored by liquid scintillation counting. A typical lower limit of detection for in-station tritium liquid samples is 1×10^{-6} µCi/gm.

Liquid Scintillation Counting

Liquid scintillation counting is a convenient, reliable, and practical way of measuring tritium in the liquid phase. The technique consists of dissolving or dispersing the tritiated compound in a liquid scintillation cocktail, and counting the light pulses emitted from the interaction between the tritium betas and the cocktail. The light pulses are counted by a pair of photomultiplier tubes which, when coupled with a discriminator circuit, can effectively distinguish between tritium betas and those from other sources.

TVA's liquid scintillation counters are periodically calibrated with radioactive sources which are traceable to national standards. The counters are checked periodically with standard radioactive sources in accordance with instrument specific calibration and maintenance procedures.

Conclusions

TVA's review of its process and effluent monitoring and sampling equipment program has determined that this program requires minor modifications for a TPC. These changes are limited to the modification of the Auxiliary Building and Shield Building Exhaust tritium sampling from periodic grab samples to continuous sampling, and other sample frequency enhancements to the existing monitoring programs. See sections 2.9.6, 2.11.3 and 2.11.4.

TVA's current techniques for tritium air sampling, liquid monitoring, and liquid scintillation counting are appropriate and modifications are not warranted.

1.5.16 Use of LOCTA-JR Code for LOCA Analyses

NUREG-1672, Section 2.15.5, "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."

Response

TVA has submitted (References 1.5.16.1 and 1.5.16.2) the LOCTA-JR code for NRC staff review. The NRC issued a SER (Reference 1.5.16.3) on January 17, 2001 documenting its acceptance of the TVA response.

References

1.5.16.1 Letter from TVA (Mark J. Burzynski) to NRC Document Control Desk dated June 23, 2000, regarding SEQUOYAH (SQN) AND WATTS BAR (WBN) NUCLEAR PLANTS - TRITIUM PROGRAM (This letter provided LOCTA_JR Proprietary Version, R0).

1.5.16.2 Letter from TVA (Mark J. Burzynski) to NRC Document Control Desk dated October 5, 2000, regarding SEQUOYAH (SQN) AND WATTS BAR (WBN) NUCLEAR PLANTS - TRITIUM PROGRAM (This letter provided LOCTA_JR Proprietary Version, R1 and the non-proprietary version of the same code).

1.5.16.3 Letter from NRC (Robert E. Martin) to TVA (J.A. Scalice) dated January 17, 2001, regarding SAFETY EVALUATION OF LOCTAJR CODE FOR LOSS -OF-COOLANT ACCIDENT ANALYSIS OF FUEL RODS - WATTS BAR NUCLEAR PLANT, UNIT 1, AND SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA9520, MA9583, MA9584).

1.5.17 ATWS Analysis

Action

NUREG-1672, Section 2.15.7, "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."

Response

TVA has submitted (Reference 1.5.17.1) the ATWS analysis for NRC staff review. The NRC issued a SER (Reference 1.5.17.2) on March 16, 2001 documenting its acceptance of the TVA response.

References

1.5.17.1 Letter from TVA (Pedro Salas) to NRC Document Control Desk dated September 29, 2000, regarding SEQUOYAH NUCLEAR PLANT (SQN) - TRITIUM PRODUCTION - ANTICIPATED TRANSIENTS WITHOUT SCRAMS (ATWS).

1.5.17.2 Letter from NRC (L. Mark Padovan) to TVA (J.A. Scalice) dated March 16, 2001, regarding SEQUOYAH UNITS 1 AND 2, AND WATTS BAR UNIT 1, RE: TRITIUM PRODUCTION PGORAM - NURGE-1672 INTERFACE ISSUE 17 - ANTICIPATED TRANSIENT WITHOUT SCRAM ANALYSES (TAC NOS. MA9583 and MB0515).

1.6 SEQUOYAH PLANT SPECIFIC CHANGES

During the NRC's review of the TPCTR, the NRC determined that a facility undertaking irradiation of a tritium production core will require changes to the Technical Specifications (TS) contained in Appendix A of any facility operating license. The evaluations and analyses for SQN contained in this report along with the TPCTR and the SER provide the technical bases for the Sequoyah TS changes necessary to irradiate TPBARs. In addition, TVA anticipates implementation of a 1.3% (from 3411 to 3455 MW_t) thermal power up-rate prior to initial irradiation of the TPBARS in Units 1 and/or 2.

1.6.1 Technical Specifications

The following TS sections were identified in the SER as candidates for change when incorporating TPBARs:

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

1.6.2 Sequoyah Specific TS Changes

TVA has evaluated the use of TPBARs in SQN Units 1 and 2 and has determined that the following TS sections require modification to support TPBAR implementation:

1. TS 3/4.5.1 – Cold Leg Accumulator – Boron Concentration Increase
2. TS 3/4.5.5 Refueling Water Storage Tank – Boron Concentration Increase
3. TS 5.3 Design Features/Reactor Core/Fuel Assemblies
4. TS 5.6 Design Features, Fuel Storage

These TS changes will be provided in a future submittal. This submittal to the NRC will request an amendment to the SQN operating license to allow operation with a tritium production core. Item 4 above and Section 1.5.10 will be submitted, if required, at a later date.

The NRC in their SER for the TPCTR identified several potential TS changes (see Section 1.6.1) that could be required to support operation with TPBARs. Two of the identified TS changes are not required for SQN. Their applicability to SQN is discussed below:

a) TS 3.4.9 (TS 3.4.3 in NUREG-1431, Rev. 1) – RCS Pressure and Temperature (P/T) Limits

It has been demonstrated that placing burnable poisons in specific peripheral assemblies suppresses the power in those assemblies. This results in a lower fluence at the maximum vessel exposure point with the tritium production core fluence projections such that the existing projections are bounding. Therefore, there will be no change to the Appendix G P/T limit curves in the TS relative to those for the 1.3% uprated core. Therefore, no change to TS 3.4.9 is required.

b) TS 3.4.12 – Low-Temperature Overpressure Protection (LTOP) System

It has been demonstrated that the 1.3% uprated core Appendix G limit curves remain applicable and, consequently, the existing LTOPS analyses and setpoints remain applicable for Sequoyah with TPBARs. Therefore, no change to TS 3.4.12 is required.

1.6.3 Thermal Power Uprate

Although the SQN thermal power up-rate of 1.3% is not required for the implementation and utilization of TPBARs, TVA anticipates implementation of a thermal power up-rate prior to initial insertion of the TPBARs into SQN Units 1 and/or 2. Hence, all evaluations and analyses contained in this report have assumed the up-rated power level of 3455 MW_t (versus the current rating of 3411 MW_t). Therefore, additional TPBAR licensing actions should not be required as a result of a future power uprate up to 1.3%.

1.7 REFERENCES

1. NDP-98-181, Revision 1, "Tritium Production Core (TPC)", Unclassified, Non-proprietary version, dated February 8, 1999, by Westinghouse Electric Company.

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

3. NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report on the Tritium Production Core", dated May 1999, by the NRC.

Table 1-1

NSSS Performance Parameters

	TPCRD	SQNREF	SQNTPC
Key Configuration Parameters			
Number of Loops	4	4	4
Reactor Coolant Pump (hp)	7000	6000	6000
17x17 Fuel Assembly Rod Array	Vantage+	Mark-BW17	Mark-BW17
Containment Type	Dry	Ice	Ice
NSSS Performance Parameters			
NSSS Power, MWt	3579	3423	3467
Reactor Power, MWt	3565	3411	3455
Thermal Design Flow, gpm/loop	93600	87000	87000
Reactor Coolant Pressure, psia	2250	2250	2250
Core Bypass Flow Fraction	8.4%	7.5%	7.5%
Reactor Coolant Temperatures, °F			
Core Outlet	625.0	616.0	616.4
Vessel Outlet (T_{hot})	620.0	611.2	611.6
Core Average	593.0	582.4	582.5
Vessel Average	588.4	578.2	578.2
Vessel/Core Inlet (T_{cold})	556.8	545.2	544.8
Steam Generator Outlet	556.5	544.9	544.5
Steam Generator Performance			
Steam Temperature, °F	538.4	518.5	517.5
Steam Pressure, psia	950	802	795
Steam Flow, million lb/hr	15.92	14.89	15.12
Feedwater Temperature, °F	446.0	434.6	436.3
SG Maximum Tube Plugging, %	10	15	15

Table 1-2

Core Design Parameters for the Sequoyah Tritium Production Cores

Design Parameters	SQNREF Typical	TPCRD Equilibrium Cycle	SQNTPC Equilibrium Cycle
Total number of feed assemblies	80 – 85	140	96
Feed loading (mtU)	31.74 – 38.62	59.2	43.66
Number of TPBARs	0	3344	2256
Total grams of tritium produced	NA	2805	2007

Table 1-3

Key Physical Parameters for Sequoyah Units

Fuel assemblies in the core	193
Number of RCCAs	53
Fuel rods per assembly	264
Available guide thimbles per assembly	24
Active length of fuel, in.	144
Active length of TPBARs, in.	132

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
1.8	Interfaces for Standard Designs	2.1	No	NA
2.1.1	Site Location and Description	2.2	No	NA
2.1.2	Exclusion Area Authority and Control	2.2	No	NA
2.1.3	Population Distribution	2.2	No	NA
2.2.1 2.2.2	Identification of Potential Hazards in Site Vicinity	2.2	No	NA
2.2.3	Evaluation of Potential Accidents	2.2	No	NA
2.3.1	Regional Climatology	2.2	No	NA
2.3.2	Local Meteorology	2.2	No	NA
2.3.3	Onsite Meteorological Measurements Programs	2.2	No	NA
2.3.4	Short Term Diffusion Estimates	2.2	No	NA
2.3.5	Long Term Diffusion Estimates	2.2	No	NA
2.4.1	Hydrologic Description	2.2	No	NA
2.4.2	Floods	2.2	No	NA
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2.2	No	NA
2.4.4	Potential Dam Failures	2.2	No	NA
2.4.5	Probable Maximum Surge and Seiche Flooding	2.2	No	NA
2.4.6	Probable Maximum Tsunami Flooding	2.2	No	NA
2.4.7	Ice Effects	2.2	No	NA
2.4.8	Cooling Water Canals and Reservoirs	2.2	No	NA
2.4.9	Channel Diversions	2.2	No	NA
2.4.10	Flooding Protection Requirements	2.2	No	NA
2.4.11	Cooling Water Supply	2.2	No	NA
2.4.12	Groundwater	2.2	No	NA
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	2.2	Yes	2.11.3
2.4.14	Technical Specifications and Emergency Operation Requirements	2.2	No	NA
2.5.1	Basic Geologic and Seismic Information	2.2	No	NA
2.5.2	Vibratory Ground Motion	2.2	No	NA
2.5.3	Surface Faulting	2.2	No	NA
2.5.4	Stability of Subsurface Materials and Foundations	2.2	No	NA
2.5.5	Stability of Slopes	2.2	No	NA
3.2.1	Seismic Classification	2.3	No	NA
3.2.2	System Quality Group Classification	2.3	No	NA
3.3.1	Wind Loadings	2.3	No	NA
3.3.2	Tornado Loadings	2.3	No	NA
3.4.1	Flood Protection	2.3	No	NA

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
3.4.2	Analysis Procedures	2.3	No	NA
3.5.1.1-3.5.1.6	Missiles	2.3	No	NA
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2.3	No	NA
3.5.3	Barrier Design Procedures	2.3	No	NA
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	2.3	No	NA
3.6.2	Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	2.3	No	NA
3.7.1	Seismic Design Parameters	2.3	No	NA
3.7.2 3.7.3	Seismic System and Subsystem Analysis	2.3	No	NA
3.7.4	Seismic Instrumentation	2.3	No	NA
3.8.1 3.8.2	Concrete Containment/Steel Containment	2.3	No	NA
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	2.3	No	NA
3.8.4	Other Seismic Category 1 Structures	2.3	No	NA
3.8.5	Foundations	2.3	No	NA
3.9.1	Special Topics for Mechanical Components	2.3	Yes	Sec. 4, Table 4-1
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	2.3	Yes	Sec. 4, Table 4-1
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	2.3	Yes	Sec. 4, Table 4-1
3.9.4	Control Rod Drive Systems	2.3	Yes	Sec. 4, Table 4-1
3.9.5	Reactor Pressure Vessel Internals	2.3	Yes	Sec. 4, Table 4-1
3.9.6	Inservice Testing of Pumps and Valves	2.3	No	NA
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	2.3	No	NA
3.11	Environmental Qualification of Mechanical and Electrical Equipment	2.3	Yes	Sec. 4, Table 4-1
4.2	Fuel System Design	2.4	Yes	2.4.2
4.3	Nuclear Design	2.4	Yes	2.4.3
4.4	Thermal and Hydraulic Design	2.4	Yes	2.4.4
4.5.1	Control Rod Drive Structural Materials	2.4	No	NA

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
4.5.2	Reactor Internal and Core Support Materials	2.4	No	NA
4.6	Functional Design of Control Rod Drive System	2.4	Yes	Sec. 4, Table 4-1
5.2.1.1 5.2.1.2	Compliance with the Codes and Standards Rule, 10CFR50.55a and Applicable Code Cases	2.5	No	NA
5.2.2	Overpressurization Protection	2.5	Yes	Sec. 4, Table 4-1
5.2.3	Reactor Coolant Pressure Boundary Materials	2.5	No	NA
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	2.5	No	NA
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	2.5	No	NA
5.3.1	Reactor Vessel Materials	2.5	Yes	1.5.4
5.3.2	Pressure-Temperature Limits	2.5	Yes	1.5.4
5.3.3	Reactor Vessel Integrity	2.5	Yes	1.5.4
5.4.1.1	Pump Flywheel Integrity (PWR)	2.5	No	NA
5.4.2.1	Steam Generator Materials	2.5	No	NA
5.4.2.2	Steam Generator Tube Inservice Inspection	2.5	No	NA
5.4.7	Residual Heat Removal (RHR) System	2.5	Yes	Sec. 4, Table 4-1
5.4.11	Pressurizer Relief Tank	2.5	No	NA
5.4.12	Reactor Coolant System High Point Vents	2.5	No	NA
6.1.1	Engineered Safety Features Materials	2.6	No	NA
6.1.2	Protective Coating Systems (Paints) – Organic Materials	2.6	Yes	Sec. 4, Table 4-1
6.2.1	Containment Functional Design	2.6	Yes	Sec. 4, Table 4-1 6.2.1
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	2.6	No	NA
6.2.1.1.B	Ice Condenser Containments	2.6	No	NA
6.2.1.2	Subcompartment Analysis	2.6	No	NA
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	2.6	Yes	Sec. 4, Table 4-1, 6.2.1

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	2.6	Yes	Sec. 4, Table 4-1, 6.2.1
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2.6	Yes	Sec. 4, Table 4-1, 6.2.1
6.2.2	Containment Heat Removal Systems	2.6	Yes	Sec. 4, Table 4-1
6.2.3	Secondary Containment Functional Design	2.6	No	NA
6.2.4	Containment Isolation System	2.6	No	NA
6.2.5	Combustible Gas Control in Containment	2.6	Yes	1.5.6
6.2.6	Containment Leakage Testing	2.6	No	NA
6.2.7	Fracture Prevention of Containment Pressure Boundary	2.6	No	NA
6.3	Emergency Core Cooling System	2.6	Yes	Sec. 4, Table 4-1
6.4	Control Room Habitability Systems	2.6	Yes	1.5.5
6.5.1	ESF Atmosphere Cleanup Systems	2.6	No	NA
6.5.2	Containment Spray as a Fission Product Cleanup System	2.6	No	NA
6.5.3	Fission Product Control Systems and Structures	2.6	Yes	Sec. 4, Table 4-1
6.5.4	Ice Condenser as a Fission Product Cleanup System	2.6	No	NA
6.6	Inservice Inspection of Class 2 and 3 Components	2.6	No	NA
7.1	Instrumentation and Controls-Introduction	2.7	No	NA
7.2	Reactor Trip System	2.7	Yes	Sec. 4, Table 4-1
7.3	Engineered Safety Features Systems	2.7	Yes	Sec. 4, Table 4-1
7.4	Systems Required for Safe Shutdown	2.7	Yes	Sec. 4, Table 4-1
7.5	Information Systems Important to Safety	2.7	Yes	Sec. 4, Table 4-1
7.6	Interlock Systems Important to Safety	2.7	No	NA
7.7	Control Systems	2.7	Yes	Sec. 4, Table 4-1
8.0	Electric Power	2.8	Yes	Sec. 4, Table 4-1
9.1.1	New Fuel Storage	2.9	Yes	1.5.10

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
9.1.2	Spent Fuel Storage	2.9	Yes	1.5.10
9.1.3	Spent Fuel Pool Cooling and Cleanup System	2.9	Yes	1.5.11
9.1.4	Light Load Handling System	2.9	Yes	1.5.7
9.1.5	Overhead Heavy Load Handling Systems	2.9	Yes	2.9.1.1
9.2.1	Station Service Water System	2.9	Yes	1.5.8
9.2.2	Reactor Auxiliary Cooling Water Systems	2.9	Yes	1.5.12
9.2.3	Demineralized Water Makeup System	2.9	Yes	1.5.13
9.2.4	Potable and Sanitary Water Systems	2.9	No	NA
9.2.5	Ultimate Heat Sink	2.9	Yes	1.5.9
9.2.6	Condensate Storage Facilities	2.9	No	NA
9.3.1	Compressed Air System	2.9	No	NA
9.3.2	Process and Post-Accident Sampling Systems	2.9	Yes	2.9.6
9.3.3	Equipment and Floor Drainage System	2.9	No	NA
9.3.4	Chemical and Volume Control System	2.9	Yes	2.9.1.2
10.0	Steam and Power Conversion System	2.10	Yes	Sec. 4, Table 4-1
11.1	Source Terms	2.11	Yes	2.11.2
11.2	Liquid Waste Management Systems	2.11	Yes	2.11.3 and 1.5.14
11.3	Gaseous Waste Management Systems	2.11	Yes	2.11.4
11.4	Solid Waste Management Systems	2.11	Yes	2.11.5
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	2.11	Yes	1.5.15
12.1	Assuring that Occupational Radiation Exposures are As Low As is Reasonably Achievable (ALARA)	2.12	No	NA
12.2	Radiation Sources	2.12	Yes	2.12.2
12.3-12.4	Radiation Protection Design Features	2.12	Yes	2.12.3
12.5	Operational Radiation Protection Program	2.12	Yes	2.12.4
13.1.1	Management and Technical Support Organization	2.13	No	NA

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
13.1.2-13.1.3	Operating Organization	2.13	No	NA
13.2.1-13.2.2	Training	2.13	Yes	2.13.1.1
13.3	Emergency Planning	2.13	Yes	2.13.1.2
13.4	Operation Review	2.13	No	NA
13.5.1-13.5.2	Administrative, Operating, and Maintenance Procedures	2.13	Yes	2.13.1.3
13.6	Physical Security	2.13	Yes	2.13.2
14.2	Initial Plant Test Program-Final Safety Analysis Report	2.14	Yes	2.14.2
15.1.1-15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	2.15	Yes	Sec. 4, Table 4-1
15.1.5	Steam System Piping Failures Inside and Outside of Containment	2.15	Yes	Sec. 4, Table 4-1
15.1.5, Appendix A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	2.15	Yes	2.15.6.4
15.2.1-15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve, and Steam Pressure Regulator Failure (Closed)	2.15	Yes	Sec. 4, Table 4-1
15.2.6	Loss of Non-emergency AC Power to the Station Auxiliaries	2.15	Yes	Sec. 4, Table 4-1
15.2.7	Loss of Normal Feedwater Flow	2.15	Yes	Sec. 4, Table 4-1
15.2.8	Feedwater System Pipe Breaks Inside and Outside of Containment	2.15	Yes	Sec. 4, Table 4-1
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	2.15	Yes	Sec. 4, Table 4-1
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2.15	Yes	Sec. 4, Table 4-1
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition	2.15	Yes	Sec. 4, Table 4-1

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2.15	Yes	Sec. 4, Table 4-1
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2.15	Yes	Sec. 4, Table 4-1
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	2.15	Yes	Sec. 4, Table 4-1
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	2.15	Yes	Sec. 4, Table 4-1
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	2.15	Yes	Sec. 4, Table 4-1
15.4.8	Spectrum of Rod Ejection Accidents	2.15	Yes	Sec. 4, Table 4-1
15.4.8, Appendix A	Radiological Consequences of a Control Rod Ejection Accident	2.15	Yes	2.15.6.7
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	2.15	Yes	Sec. 4, Table 4-1
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	2.15	Yes	Sec. 4, Table 4-1
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2.15	Yes	2.15.6.9
15.6.3	Radiological Consequences of Steam Generator Tube Failure	2.15	Yes	2.15.6.5
15.6.5 and Appen-dices A & B	Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	2.15	Yes	2.15.5 and 2.15.6.3
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	2.15	Yes	2.11.3
15.7.4	Radiological Consequences of Fuel Handling Accidents	2.15	Yes	2.15.6.6
15.7.5	Spent Fuel Cask Drop Accidents	2.15	Yes	Sec. 4, Table 4-1
15.8	Anticipated Transients Without Scram (ATWS)	2.15	Yes	1.5.17

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations (Continued)

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Sequoyah Report Section
16.0	Technical Specifications	2.16	Yes	Enclosure 1 to LAR Submittal and Sec. 1.6
17.1	Quality Assurance During the Design and Construction Phases	2.17	Yes	1.5.2, 2.17
17.2	Quality Assurance During the Operations Phase	2.17	Yes	1.5.2, 2.17
17.3	Quality Assurance Program Description	2.17	No	NA
18.1	Control Room	2.18	No	NA
18.2	Safety Parameters Display System (SPDS)	2.18	No	NA

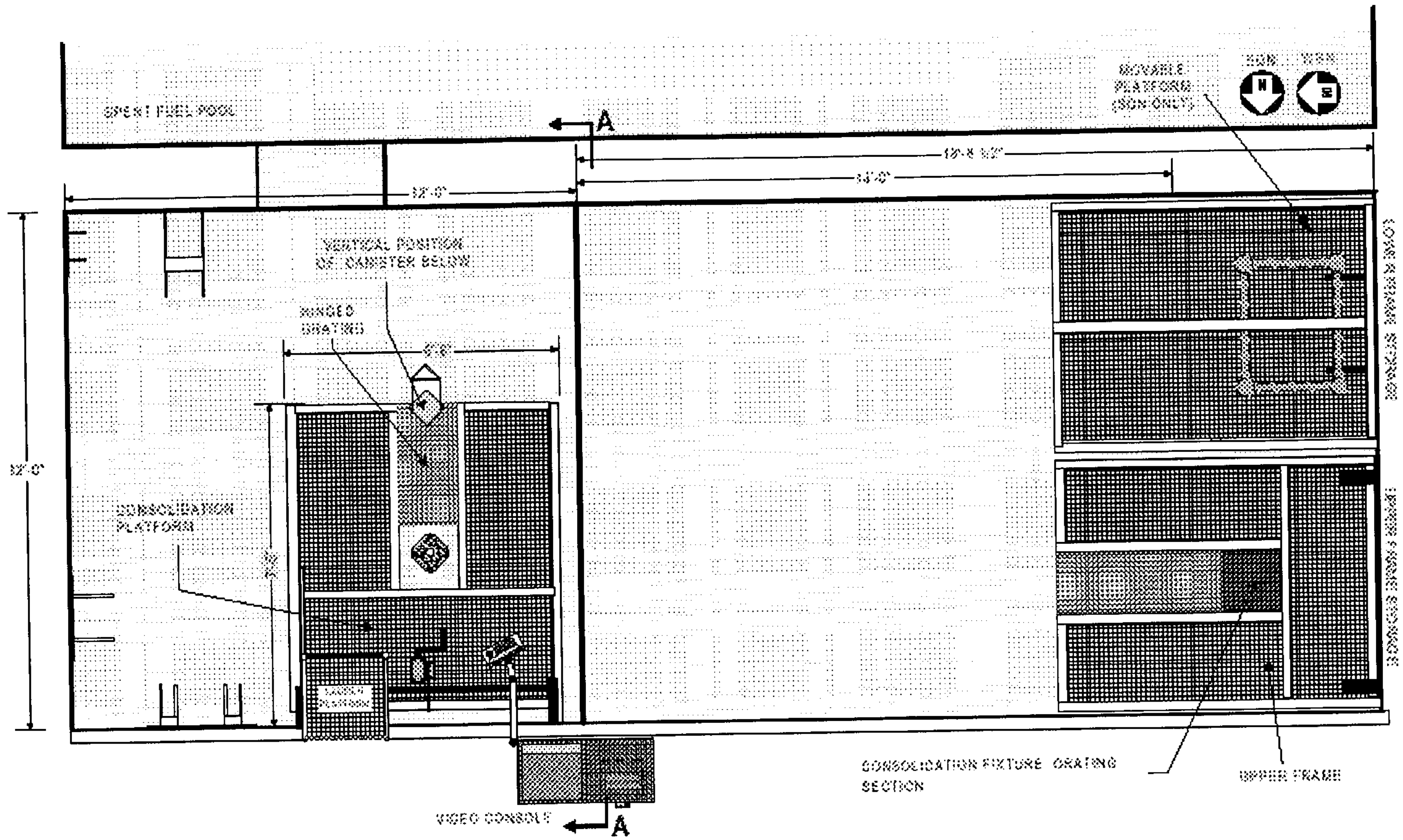


Figure 1.5.1-1

Consolidation Plan

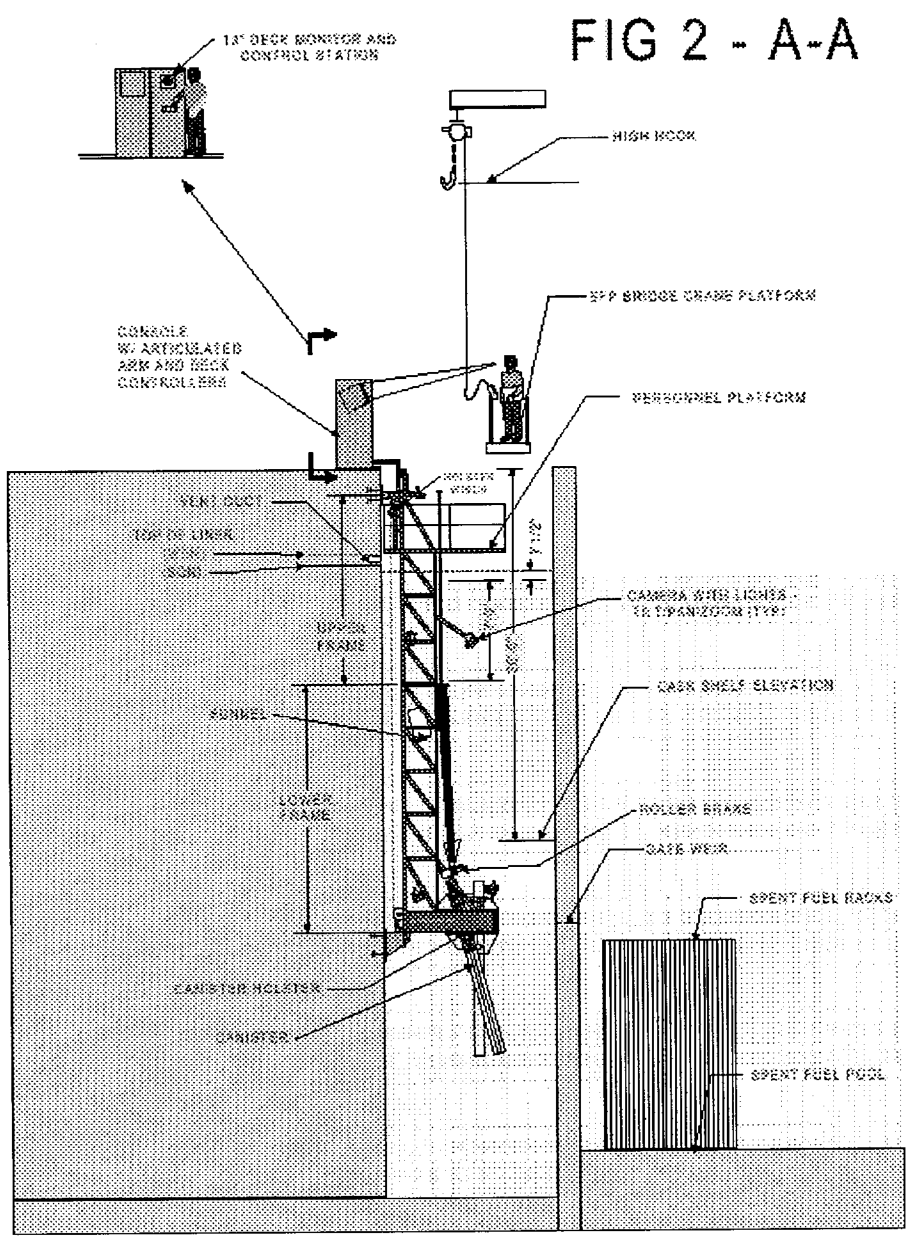
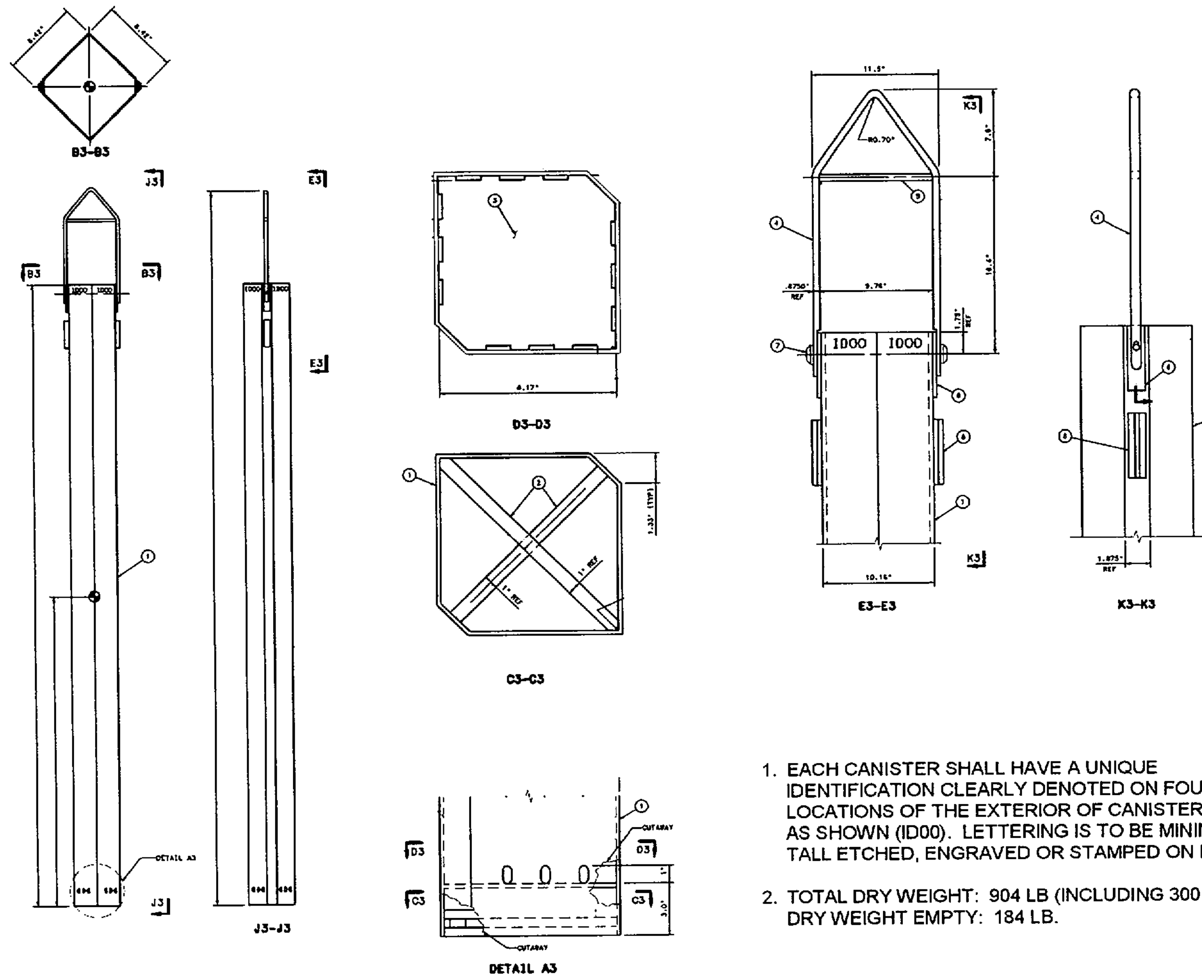


Figure 1.5.1-2

Consolidation Plan A-A



1. EACH CANISTER SHALL HAVE A UNIQUE IDENTIFICATION CLEARLY DENOTED ON FOUR LOCATIONS OF THE EXTERIOR OF CANISTER AT TOP AS SHOWN (ID00). LETTERING IS TO BE MINIMUM ¾" TALL ETCHED, ENGRAVED OR STAMPED ON METAL.
2. TOTAL DRY WEIGHT: 904 LB (INCLUDING 300 TP BARS). DRY WEIGHT EMPTY: 184 LB.

ITEM		NAME
1		CANISTER
2		ENERGY TRANSFER BARS
3		CANISTER BOTTOM PLATE
4		BAIL
5		BAIL CROSS MEMBER
6		HINGE PLATE
7		HINGE PIN
8		LIFTING LUG

Figure 1.5.1-3

Consolidation Canister

	H	G	F	E	D	C	B	A
8								
9								
10								
11								
12								
13							B13 24 BPRAs	
14						C14 24 BPRAs		
15								

Figure 1.5.4-1

Location of BPR Assemblies used for Suppressing Neutron Fluence on Sequoyah Vessel Wall in Example Equilibrium Cycle

SECTION 2 STANDARD REVIEW PLAN EVALUATION

2.2.1 Accidental Releases of Liquid Effluents Evaluation

This evaluation is addressed in Section 2.11.3.

2.4 REACTOR

2.4.2 Fuel Design Evaluation

<u>Fuel Assembly Structural Integrity</u>

Introduction

The FRA-ANP Mark-BW fuel assembly was evaluated to determine the impact of the TPBAR on the fuel assembly structural integrity. The fuel mechanical design was assessed in accordance with the guidelines in Section 4.2 of the Standard Review Plan. Only the weight of the fuel assembly containing 24 TPBARs has changed with respect to the reference fuel assembly configuration and from previous SRP required analyses.

Methodology

A comparison was performed to evaluate the impact of the additional weight of each fuel assembly on the grid load margin available for the SQN plant in the Mark-BW fuel assembly structural analysis. The structural adequacy of the Mark-BW fuel assembly design was evaluated using NRC requirements for combined seismic and LOCA loads per Appendix A to SRP 4.2 and approved methodology (Reference 1). The grid load results for the 17x17 Mark-BW fuel assembly design were reviewed. The combined seismic and LOCA grid load is considerably less than the allowable grid strength, resulting in sufficient grid load margin for the SQN plants, based on a very conservative analysis incorporating the TPBAR.

Input Parameters and Assumptions

The nominal weight for each TPBAR is 2.3 lbs. Therefore, the additional weight per assembly totals approximately 63 lbs for 24 TPBARs. This is approximately 4% of the Mark-BW fuel assembly's weight. A conservative weight of 70 lbs was used in the analysis.

Results

Because the TPBAR assembly is a hanging structure supported by the top nozzle adapter plate of the fuel assembly and the rodlets are hanging in the guide thimble tubes, the added weight can be considered to be part of the fuel assembly nozzle support. However, for the evaluation, the TPBAR weight was conservatively assumed to be distributed along the length of the fuel assembly. The rodlet stiffness was not considered in the analysis for conservatism to maximize the fuel assembly frequency change. The TPBAR assembly weight was shown to have a minimal effect on the fuel assembly dynamic characteristics. Therefore, the TPBAR design for the SQN plants impose no significant impact to the fuel assembly structural integrity evaluation.

Conclusions

The grid load margin for the SQNTPC was assessed. With a conservative modeling of the mass and stiffness effects of the TPBAR, there is still more than sufficient grid load margin. The use of the TPBAR assemblies in the SQN plants has only a small effect on the Mark-BW fuel assembly faulted condition structural loads. Changes to the dynamic characteristics of the fuel assembly are minimal. In addition, interactions between the TPBARs and guide tubes would 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 TPBAR is negligible. These factors would serve to further reduce the impact of the added weight of the TPBAR assemblies on the LOCA/seismic analysis for SQNTPC. The supplemental faulted condition evaluation is specific to FRA-ANP fuel and faulted condition methodology approved by the NRC in Reference 1.

Fuel Rod Design

The FRA-ANP fuel rod design methods are given in Reference 2. FRA-ANP Mark-BW fuel rod designs are approved for use up to a rod average burnup level of 60 GWd/mtU. The NRC approved TACO3 code (Reference 3) was used to simulate in-reactor behavior of the fuel rods.

The important areas of fuel rod mechanical performance are cladding stress, cladding fatigue, cladding strain, cladding creep collapse, cladding corrosion, and fuel rod growth. The cladding stress and fatigue analyses retain large margins and are insensitive to the introduction of TPBARS in future cycles. The fuel rod growth evaluation is also insensitive to the introduction of TPBARS. The cladding corrosion analysis is evaluated on a cycle specific basis for the SQN reactors. Comparisons of fuel rod power histories and operating parameters between cycles using TPBARS and those cycles without indicate that similar margins to cladding corrosion limits will be maintained.

The effect of the use of TPBARS on fuel rod behavior was evaluated in the areas of cladding strain and creep collapse. For the cladding strain evaluation, the generic Mark-BW fuel rod cladding transient strain limits were shown to be valid for use in the TPBAR cycles. Also, present fuel rod cladding creep collapse lifetimes for the Mark-BW fuel rod design were maintained for the TPBAR cycles.

Conclusions

Since adequate fuel rod performance margins exist, the existing fuel rod design is valid for SQNTPC.

References

1. Mark-C Fuel Assembly LOCA-Seismic Analyses, BAW-10133P-A, Rev. 1, Framatome Cogema Fuels, Lynchburg, Virginia, June 1986.

2. Extended Burnup Evaluation, BAW-10186P-A, Rev. 1, Framatome Cogema Fuels, Lynchburg, Virginia, April 2000.

3. TACO3 Fuel Pin Thermal Analysis Code, BAW-10162P-A, B&W Fuel Company, Lynchburg, Virginia, October 1989.

2.4.3 Nuclear Design

Introduction

Conceptual core designs were developed and analyzed for the SQNTPCs. This section describes the nuclear design methodology, design bases, core design descriptions, core power distribution and reactivity coefficient evaluations, and effects of extended shutdowns for the representative SQNTPCs.

First and equilibrium cycle core designs were developed for the SQNTPCs using feed batch sizes of 96 fuel assemblies. The overall goal of these core designs and associated analyses was to determine the feasibility of producing tritium with a batch size larger than current SQN reload cores. The design inputs and criteria applied to SQNREF core designs were applied to the SQNTPCs. The cycle energy chosen was 510 effective full power days (EFPD) at a rated thermal power of 3455 MWt, which includes 10 EFPD of power coastdown; the corresponding cycle burnup was about 21,100 MWd/mtU.

The Tritium Producing Burnable Absorber Rods (TPBARs) function in the reactor core in a manner similar to the burnable poison (BP) rods that have been used in recent SQN core designs. The primary design goal for these core designs was to produce the largest quantity of tritium possible. With few exceptions, enrichments of 4.95 w/o were used to achieve this objective; exceptions were necessary in the first transition cycle to achieve better power peaking control. Between 12 and 24 TPBARS were used in each TPBAR assembly; the first transition cycle used fewer TPBARs due to cycle energy requirements.

Table 2.4.3-1 lists SQNTPC operating parameters and design objectives. Both the first and equilibrium cycles use the same type of fuel, the Mark-BW fuel assembly with Zircaloy-4 grids and cladding. This is the same type of fuel currently employed in both Sequoyah units.

The cores were designed to meet established design and safety limits such as peaking limits of:

- an $F_Q(X,Y,Z)$ * P ECCS limit = 2.50 * K(Z), and
- a design $F_{\Delta H}(X,Y)$ limit = 1.70.

The moderator temperature coefficient Technical Specification limit at hot zero power (HZP) is <0 pcm/°F. The shutdown margin (SDM) limit is 1.6 %Δk/k. A comprehensive set of nuclear analyses was performed for these cores in which all applicable safety parameters were calculated and compared to values in the

SQN safety analysis bases. The approved methodology to do this is described in Reference 1. With four notable exceptions, all key safety parameters for these cores fall within the ranges that are typically assumed for the SQN Units. The exceptions (discussed below) are shutdown margin (resulting in the relocation of four RCCAs), Doppler Only Power Coefficient (DOPC) at zero power, HZP ejected rod worth at BOC (which may affect the rod insertion limits), and post-LOCA recriticality (which affects the RWST boron concentration and cold leg accumulator boron concentration). These exceptions will be addressed by making necessary changes to the SQN units' control rod pattern and Technical Specifications.

With the primary objective of maximizing the production of tritium in each core, TPBARs are loaded primarily in the feed batch assemblies. In the equilibrium core, a few feed batch assemblies that are located in control rod locations do not contain TPBARs, and conversely some TPBARs are loaded into once-burned fuel assemblies. This was done primarily to obtain better power peaking control. The TPBAR design is similar to the design used in the TPCTR and the Watts Bar Lead Test Assemblies (LTAs) topical report (Reference 2); however, two ^{6}Li linear loadings are used, 0.029 and 0.032 gm/in. The dual concentrations provide some additional core design flexibility for power distribution control. The poison length of the TPBARs used in the SQNTPC is 132 inches.

Burnable Poison Rod Assemblies (BPRAs) containing 3.5 w/o B_4C in Al_2O_3 pellets were used on the core periphery for vessel fluence control in the equilibrium fuel cycle. This practice was necessitated because of 1) the reduced burnup of fuel assemblies located on the core periphery that result from the larger feed batch sizes, 2) the higher fuel enrichments, and 3) the interior TPBARs that push more power to peripheral core locations.

Gadolinia-urania (Gd_2O_3-UO_2) pellets were used as an integral burnable absorber in a portion of the fuel rods. Typically, up to 24 gadolinia rods with gadolinia concentrations between 2 and 8 w/o are arranged in the fuel assembly for power peaking and soluble boron control. The fuel enrichment in the gadolinia fuel pellets is slightly reduced compared to the uranium fuel pellet enrichment. The gadolinia-urania and ^{6}Li pellet stacks are the same length and are both vertically centered. The use of gadolinia in core designs is consistent with current practice at SQN. The active absorber stack length has been increased from 126 to 132 inches as a result of discontinuation of axial blankets in the SQNTPCs. The active fuel region above and below the gadolinia pellets in feed assemblies for these core designs are natural uranium pellets.

Most of the 96 fuel assemblies comprising each feed batch contain a primary enrichment of 4.95 w/o ^{235}U. The exceptions are eight fuel assemblies with reduced uranium enrichments in the transition core for improved power distribution control. Except for the reduced enrichment in the gadolinia rods, no zone loading or axial blankets are employed in the feed batches. Burned fuel in the transition cycle reflects a transition from a typical SQN fuel cycle that contains burned fuel with both low-enriched axial blankets and gadolinia rods.

Conclusions

The differences as compared to the TPCTR are primarily due to the lower feed batch sizes used in the SQNTPC fuel cycles and the different fuel management practices at SQN. The significant differences are as follows.

1. A feed batch of 96 Mark-BW fuel assemblies was used instead of 193 and 140 VANTAGE+™ fuel assemblies.
2. Two ^{6}Li concentrations were used instead of one; concentrations slightly higher (0.032 gm/in) and lower (0.029 gm/in) than that in the TPCTR analysis (0.030 gm/in) were used.
3. A singular, longer ^{6}Li poison column length of 132 inches, centered with respect to the fuel stack was used. The TPCTR analysis used 127.5 and 128.5 inch lengths, and the Watts Bar LTAs used a 142 inch length.
4. Gadolinia (Gd_2O_3) was used as integral burnable absorber instead of IFBA (ZrB_2); fuel enrichment was slightly reduced in the fuel pellets that contain gadolinia.
5. Burnable Poison Rod Assemblies (BPRAs) containing B_4C-Al_2O_3 pellets were used on the periphery for fluence control in the equilibrium fuel cycle instead of TPBARs.
6. As few as 12 TPBARs on a single cluster were used in the transition cycle whereas no fewer than 20 per cluster were used in the TPCTR analysis.
7. No fuel rod enrichment zone loading was employed except for fuel rods containing gadolinia.

Methodology

The key neutronics codes used to perform power distribution analyses are CASMO-3 (Reference 3) and NEMO (Reference 4). NEMO solves the nodal balance equation in three dimensions to yield neutron flux, power, and reactivity. The nodal expansion method calculates nodal fluxes and currents. Discontinuity factors provide continuity of the heterogeneous fluxes at the node surfaces. Axial fuel heterogeneity is treated by setting axial node boundaries between the heterogeneities. Fuel assembly rod powers are individually calculated via the pin power reconstruction method. NEMO uses a two-group microscopic depletion model that accounts for over 20 different isotopes, including a special treatment for those isotopes that are not individually treated. Microscopic cross sections are interpolated against variables that include burnup, boron concentration, moderator specific volume, and others. The major characteristics of the NEMO model include:

- Three-dimensional, quarter-core geometry;
- Pin-by-pin power representation for each assembly;
- Thermal-hydraulic feedback.

CASMO-3 is a two-dimensional multi-group transport theory code for burnup calculations on BWR and PWR fuel assemblies or simple fuel pin cells. The code models a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowances for fuel rods loaded with gadolinia, burnable absorber rods, cluster control rods, in-core instrument channels, and water gaps. CASMO-3 provides two-group cross-sections and other data for tablesets used by the NEMO code. CASMO-3 is routinely used to calculate microscopic two-group constants for absorber pins similar to TPBARs such as burnable poison rod assemblies (BPRAs) and Pyrex burnable absorbers.

The CASMO3-NEMO code package was subjected to an extensive verification program that quantified the uncertainties associated with the use of these codes. The NRC has approved application of the CASMO3-NEMO code package for nuclear design activities (Reference 4).

For application to TPBARs, the CASMO-3 code did not require modification; however, cross-section data were added to the library for neutronic modeling to enable the depletion of TPBARs. Specifically, the isotopes of ^{6}Li, ^{7}Li, ^{3}He, and ^{3}H were added from the ENDF-B/V library. CASMO-3 results using the additional isotopes were verified using Monte Carlo Neutron Photon (MCNP) calculations to enable the modeling of the TPBARs.

The generation of cross-section and pin power libraries for NEMO is automated. For SQNTPC analysis, the cross-section generation process was modified to treat tritium, helium, and lithium isotopes. The modified process models multiple mixtures in a burnable absorber pin that allows the non-classified TPBAR model provided by PNNL to be analyzed.

The NEMO code was modified to include additional capabilities required to analyze TPBARs. The new isotopes and depletion chains were added by using existing NEMO input features. A model was developed that accounts for ^{3}He migration from the lithium absorber to the free gas region of the TPBAR including the plenum regions. NEMO will treat both a ^{3}He region within the TPBAR absorber and a ^{3}He region defined by the free gas regions. The model also allows fitting of the ^{6}Li cross sections within the TPBAR as a function of the burnup accumulated while the TPBAR is inserted. No modifications were made to the basic NEMO algorithms. The revised model allows fresh TPBARs and BPRAs to be modeled in a burned fuel assembly. Additional editing capabilities were added to NEMO to edit the isotopic concentrations of the TPBARs on pin-by-pin and nodal bases.

Conclusions

The differences between FRA-ANP methodology and that described in the TPCTR are small. The NEMO code uses two-group microscopic cross-sections versus macroscopic cross-sections. Consequently, slight changes were required for NEMO. The changes are the ability to edit data, model fresh TPBARs in burned fuel, and provide a microscopic cross-section based ^{3}He model. The NEMO ^{3}He model is different from that used in the DOE TPC topical because:

- The plenum regions may be modeled,

- ^{3}He axial redistribution with burnup is considered by independent tracking of ^{3}He in the lithium pellet matrix and the ^{3}He in the free gas regions,
- Transmutation of ^{3}He back to tritium with neutron absorption is considered, and
- The fast flux is used in the migration rate of ^{3}He from the Li pellet to free gas volume regions since the ^{3}He becomes mobile when the neutrons impart energy to the ^{3}He.

Design Bases

The design bases and functional requirements used in the nuclear design of the fuel and reactivity control systems for the SQNTPC designs are the same as those currently used in SQN fuel cycles except for the following.

- The control rod pattern will be changed as described below.
- The minimum RWST concentration will be increased (see Section 2.15.5).
- The minimum Cold Leg Accumulator concentration will be increased (see Section 2.15.5).

The design bases and functional requirements are discussed in detail in Section 4.5.3 of Reference 1. This information is applicable to the SQNTPC designs. A discussion of the design bases and the relationship to TPBARs and the SQNTPC designs are provided below.

Fuel Burnup

A limitation on initial installed excess reactivity or average discharge burnup is not required other than as is quantified in terms of other designs bases, such as core negative reactivity feedback and shutdown margin.

Due to the 96 assembly feed batch size, the discharge burnups will be slightly lower than current Sequoyah fuel cycle designs and higher than those reported in the TPCTR. The SQNTPC equilibrium cycle average discharge burnup of about 40,000 MWd/mtU is lower than those for current Sequoyah fuel cycles (45,000 MWd/mtU) and higher than those in the TPCTR designs.

Negative Reactivity Feedbacks (Reactivity Coefficients)

The design basis for SQN specifies that the Doppler coefficient will be negative and the moderator coefficient will be non-positive at power levels equal to or greater than 0% rated thermal power (RTP).

For the SQNTPC, the Doppler feedback was always negative and similar to that of the current SQNREF cores. The moderator temperature coefficients for the SQNTPCs met the requirements described above. Lower boron worth associated with the TPBAR cores helps to create a more negative moderator temperature coefficient. In general, the SQNTPC designs have more negative moderator temperature coefficients throughout core life, with one exception. The first transition TPC moderator temperature coefficient was more negative throughout core life except for cycle average burnups between 1000-2000

MWd/mtU, when the moderator temperature coefficient increased briefly and became similar to that of the SQNREF core. At MOL and EOL, the moderator temperature coefficient remained more negative than that of the SQNREF core. The total power coefficient was always negative at all power levels. The most negative Doppler Only Power Coefficient (DOPC) at zero power was outside current limits; however, evaluation of this parameter resulted in no adverse impact on safety limits or margins (see Reactivity Coefficients discussion, below).

Based on the observed feedback characteristics of the SQNTPC designs, all design bases and limits associated with reactivity feedback parameters are satisfied.

Control of Power Distribution

The design bases for core power distribution control for the SQN Units 1 and 2 are summarized as follows. These design bases apply with at least a 95% probability and 95% confidence level:

- The maximum linear heat rate will not exceed the design limit based on centerline fuel melt for both Condition I and II operation, including the maximum design overpower condition;
- The maximum linear heat rate will not exceed the design limit based on transient cladding strain criteria for both Condition I and II operation, including the maximum design overpower condition;
- The power distribution will be limited during Condition I and II operation, including the maximum design overpower condition, such that departure from nucleate boiling (DNB) does not occur, based on the approved design limit DNB ratio (DNBR);
- The maximum linear heat rate under normal operating conditions (Condition I) will not exceed the $F_Q(x,y,z) * K(z)$ limit, which comprises the initial conditions of the LOCA analysis.

Limiting core power distributions for the SQNTPC designs were evaluated using NRC-approved methods (Reference 5) to ensure that the design bases were met. Operation at the limits of Condition I was analyzed to demonstrate that the SQNTPCs would operate with acceptable margins to the F_Q and $F_{\Delta H}$ peaking limits. Condition II power distributions were analyzed to demonstrate that the SQNTPCs would also operate with acceptable margins to the core safety limits.

Maximum Controlled Reactivity Insertion Rate

The TPCTR addresses the requirements for maximum reactivity insertion rate due to withdrawal of RCCAs at power and by boron dilution. The standard reload methodology used for current Sequoyah cores was used to evaluate the SQNTPC cores. For SQNREF (see Table 2.4.3-1), the maximum control rod speed is 45 in/min. This control rod speed is the same as that used in the TPCTR for the TPCRD.

The reactivity change rates were conservatively calculated, assuming more severe axial power distributions than those allowed by core operating limits. The SQNTPC designs met all requirements imposed on the SQNREF (see Table 2.4.3-1) in terms of reactivity insertion rates. This is consistent with the results presented in the TPCTR, i.e., the TPBARs had no impact.

To ensure that the reactor can be brought to a shutdown condition following a large break LOCA, the Refueling Water Storage Tank (RWST) boron concentration will be raised to a minimum of 3600 ppm. This is necessary because of: (1) the lower worth of boron in tritium production cores relative to conventional cores, and (2) the relatively low minimum boron concentration of the ice in the ice containment (1800 ppm). The ice boron concentration, which will not be increased, is significantly smaller than the post-LOCA subcriticality sump boron requirement. Consequently, the RWST concentration must be raised higher to compensate. A minimum RWST boron concentration of 3600 ppm will ensure post-LOCA subcriticality for the SQNTPC designs.

Shutdown Margins

Minimum shutdown margin requirements are specified in the Technical Specifications for all Modes, 1 through 6. Shutdown margins were evaluated for all Modes using approved methods. The minimum required shutdown margin was found acceptable for all Modes for the SQNTPC designs. The shutdown margin evaluation for Modes 1-5 assumed the highest worth RCCA was stuck in the fully withdrawn position.

Stability

The design bases for xenon stability are that the core must be stable with respect to axial xenon oscillations, or a means to detect and suppress the oscillations must be available. Axial xenon stability was evaluated for the 96-feed transition and equilibrium fuel cycles. As a precaution, plant procedures are in place at SQN to detect and suppress an oscillation prior to exceeding any core safety limit. Xenon stability for current SQN reload cores is evaluated by calculating a stability index for simulated xenon transients at several times in cycle life. The stability index for the SQNTPCs was bounded by the values calculated for standard reload cores, i.e., the 96-feed tritium production cores were more stable, and xenon oscillations were naturally convergent.

Conclusions

The differences between FRA-ANP methodology and those described in the TPCTR are small. The NEMO code calculates three-dimensional reactivity deficits and coefficients. This fact does not adversely impact the general trends established in either report. In fact, the evaluation of the SQNTPC designs shows very similar trends to those established in the TPCTR. Based on the observed feedback characteristics of the SQNTPC designs, all design bases and limits associated with negative reactivity feedbacks, maximum reactivity insertion rates, and shutdown margins are satisfied.

Due to the use of the 96 fuel assembly feed batch size, the discharge burnups will be slightly lower than those in current Sequoyah fuel cycle designs and higher than those in the TPCTR. The SQNTPC equilibrium cycle has an average discharge burnup of about 40,000 MWd/mtU, which is lower than that in SQNREF fuel cycles (45,000 MWd/mtU) and higher than that in the TPCTR designs (30,000 MWd/mtU).

Except as noted above, the SQNREF design bases are applicable to the SQNTPC designs. The following sections describe the first and equilibrium cycle SQNTPC designs and characterize their performance in terms of typical reactivity feedbacks and shutdown margins.

Core Design Descriptions

First Cycle SQNTPC Design Description

For the first transition cycle, a total of 1360 TPBARs and 1760 gadolinia pins were used. Gadolinia patterns of 16 and 20 pins with 4 w/o Gd_2O_3 and 16 pins with 6 w/o Gd_2O_3 were used. The fuel enrichment of the gadolinia rods was reduced slightly to allow for a lower power production in the gadolinia rods consistent with current practice.

The core loading pattern for the first transition cycle consisted of a split feed batch of 88 fuel assemblies at 4.95 w/o and eight fuel assemblies at 4.75 w/o ^{235}U. The RCCA locations shown in Figure 2.4.3-3 reflect the revised SQN control rod arrangement. Four RCCAs that were previously located in peripheral core locations B12, M14, P04, and D02 were moved inward to core locations E11, L11, L05, and E05, respectively. This change was made to satisfy the SQN shutdown margin requirements of 1.6% $\Delta k/k$ while not compromising the amount of tritium production. Shutdown margin is improved in current SQN fuel cycles by placing large numbers of feed assemblies in control rod locations; however, this practice in the SQNTPCs would affect operating margins adversely and reduce tritium production because most TPBARs would then reside in burned fuel assemblies.

The TPBARs employed in this design have a ^6Li absorber length of 132 inches (cold) and are centered with respect to the active fuel stack. The gadolinia pellet stack is also 132 inches and vertically centered. In the transition cycle clusters of 12, 16, and 24 TPBARs are used; dual ^6Li loadings of 0.029 and 0.032 grams per inch are used but only one ^6Li loading is used per cluster. The axial length and position, the number of TPBARs per cluster, and the TPBAR ^6Li loadings used in this analysis should be considered as representative and among the parameters at the core designer's discretion to modify as necessary to achieve tritium production, design margin, and energy production goals.

The secondary source clusters will be placed in core locations H03 and H13, as is current practice, and will not have TPBARs. Primary source rods will not be required.

Equilibrium SQNTPC Design Description

Table 2.4.3-5 shows the fuel region description for the SQNTPC equilibrium fuel cycle design. In this design, 96 once-burned fuel assemblies and one twice-burned fuel assembly are used in conjunction with a feed batch of 96 feed assemblies. A total of 2256 TPBARs and 1520 gadolinia pins were used. Gadolinia patterns of 16 pins with 4 w/o Gd_2O_3, and 12 and 16 pins with 8 w/o Gd_2O_3 were used.

The TPBARs employed in the equilibrium core design have a ^{6}Li absorber length of 132 inches (cold) and are centered with respect to the active fuel stack. The gadolinia pellet stack is also 132 inches and vertically centered. Clusters of 20 and 24 TPBARs are used in the equilibrium cycle; dual ^{6}Li loadings of 0.029 and 0.032 grams per inch are used but only one ^{6}Li loading is used per cluster.

Figure 2.4.3-4 shows the core loading pattern (quarter-core symmetric) for the equilibrium fuel cycle design. As in the transition cycle, two ^{6}Li loadings are used and the length and axial position remain the same. Again, the secondary sources will not contain TPBARs.

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

Conclusions

The differences as compared to the TPCTR for the first cycle and equilibrium SQNTPC designs are primarily due to the lower feed sizes used in these Sequoyah fuel cycles and the different fuel management practices at Sequoyah. The significant differences are as follows:

1. Fewer TPBARs were used due to the smaller feed batch size.
2. Gadolinia was used instead of IFBA as the integral burnable absorber.
3. Two enrichments (4.75 and 4.95 w/o ^{235}U) were used in the first cycle design. A single maximum enrichment of 4.95 w/o ^{235}U was used for all uranium fuel rods in the equilibrium cycles.
4. No enrichment zoning within the fuel assembly was used except for the reduced enrichment in the gadolinia rods.
5. The TPBARs use a slightly longer, axially centered absorber length of 132 inches.
6. Secondary source clusters did not include TPBARs.
7. More than one ^{6}Li loading was used in both the first and equilibrium cycle designs for improved power distribution control. In addition, there was a larger variation in the number of TPBARs per cluster for the first cycle transition.

<u>Nuclear Design Parameter Comparison</u>

The TPCTR provides detailed comparisons of nuclear parameters between TPCs and non-TPCs. In general, the trends observed in the TPCTR were observed in the SQNTPCs.

Conclusions

No significant differences were observed between the general trends of nuclear parameters demonstrated in the TPCTR and those observed for the SQNTPC designs.

<u>Tritium Production</u>

The maximum allowed tritium concentration defined by PNNL is 1.2 g-^{3}H/rod and is based on TPBAR pressure limitations (Reference 6). The minimum allowed tritium concentration is 0.15 g-^{3}H/rod and is based on cladding creep collapse criterion. The maximum limit must be reduced and the minimum limit increased to allow for uncertainties and operational flexibility. Components of uncertainties and operational flexibility include the integrated effects of:

- quadrant power tilt (local and global),
- effects of gadolinia manufacturing tolerances on local and global tritium production,
- effects of fuel assembly manufacturing tolerances on local and global tritium production,
- effects of TPBAR manufacturing tolerances on local and global tritium production,
- CASMO-3 versus NEMO differences in pin power reconstruction and the integrated effect on tritium production,
- cycle N-1 length flexibility, and
- power level uncertainty.

The uncertainty factors were conservatively applied to produce a total uncertainty for use in the licensing analysis. The analysis performed for the topical report amendment does not preclude future analyses that may combine factors statistically provided they are statistically independent.

During the fuel cycle design the pin-by-pin tritium concentrations were verified not to exceed the design limit with uncertainty applied. All designs evaluated met this criterion on a pin-by-pin basis.

Table 2.4.3-7 provides a summary of tritium production for the SQNTPC first transition and equilibrium cycles. The first transition cycle produced 1248 grams while the equilibrium cycle produced about 2007 grams. The average production of tritium per TPBAR was 0.918 grams in the first transition cycle and 0.889 grams in the equilibrium cycle. The maximum tritium production without uncertainty applied was 1.026 grams in the first transition cycle and 1.009 grams in the equilibrium cycle. The minimum tritium production without uncertainty applied was 0.555 grams in the first transition cycle and 0.455 grams in the equilibrium cycle. After application of uncertainties to both the maximum and minimum production, tritium production remained within the TPBAR design limits of 1.2 and 0.15 grams, respectively.

Conclusions

Due to the significantly smaller feed batch sizes used in these designs relative to the initial TPCTR analysis (50% and 69% of the original feed batch sizes), these designs produce about 44% and 72% of the initial and equilibrium cores' tritium production, respectively. However, the average tritium produced in each TPBAR is about 6 to 7% larger in the SQN designs, primarily as a result of the elimination of TPBARs in peripheral core locations.

<u>Design Variations</u>

As in the TPCTR analysis, the designs presented here should be considered representative. The primary design goal was to produce as much tritium as possible while meeting cycle energy goals and a feed batch size of 96 fuel assemblies. Other fuel design options, such as enriched axial blankets, are not precluded by the use of TPBARs but may require slightly different ^{6}Li loadings or axial configurations.

Power Distributions

Limiting Condition I and Condition II core power distributions for SQNTPC designs were calculated using the NRC-approved methods described in Reference 5. Calculations were performed for both the 96 feed transition and equilibrium fuel cycles. The simulated power distributions included the effects of transient xenon and regulating rod repositioning, and included operation at design overpower. Augmentation factors to account for modeling simplifications and uncertainties were applied as described in Reference 5. Peaking margins for each simulated power distribution were calculated relative to the core power distribution limits based on the design bases summarized above. These calculations were used to evaluate the acceptability of the TPC core designs with respect to the $f_1(\Delta I)$ and $f_2(\Delta I)$ trip reset functions and the operational axial flux difference (AFD) limits relative to SQNREF reload fuel cycles that operate with FRA-ANP fuel. The results of these calculations indicate that both the transition and equilibrium SQNTPC cores will operate with $f_1(\Delta I)$ and $f_2(\Delta I)$ trip reset function breakpoints and slopes, and AFD limits similar to those specified for reload fuel cycles using fuel designs, burnable absorber designs, and fuel management currently in use at the Sequoyah units.

Increased power peaking is caused by axial gaps between the TPBAR absorber pellet stacks at the interfaces between individual pencils (see Section 3.7.2 and Reference 7). The effect of the increase in peaking due to the gaps was accommodated explicitly in the power distribution evaluations. Conservative augmentation factors were defined and applied to the limiting power peaking factors when peaking margins were calculated. These augmentation factors were applied in addition to the standard augmentation factors used in the design and analysis of SQNREF reload cycles.

During its review of the TPCTR, the NRC staff identified compliance with the DNB criterion as an interface issue (see section 1.5.3) for which plant-specific information would be required in the licensee's submittal to support an amendment to the facility operating license for authorization to operate a tritium production core. The acceptability of the limiting core power distributions with respect to DNB performance was explicitly evaluated for the 96-feed maximum TPBAR transition and equilibrium fuel cycles. The evaluation was performed using the standard approved reload analytical methods described in Reference 5. The results of the evaluation confirmed that the presence of TPBARs can be accommodated, at the power uprate condition of 3455 MWt, without violation of the DNB design bases. Therefore, the presence of TPBARs in the reload core design did not challenge the DNB criterion. An explicit check of the DNB criterion is included in the cycle-specific reload safety evaluation performed for each Sequoyah reload core. Continued performance of this check will validate the acceptability of each reload core for operation within the DNB design limits.

In summary, the core power distribution evaluations performed for 96-feed maximum TPBAR transition and equilibrium cycles demonstrated that SQNTPCs can operate at the uprated thermal power of 3455 MWt without violation of any of the nuclear design bases. NRC-approved methodology was used to perform these evaluations. The resulting core protective and operating limits were typical of those established for current standard SQN reload cores operating with FRA-ANP fuel. Preservation of the DNB criterion was confirmed for operation within the bounds of Conditions I and II, including operation at design overpower.

Conclusions

Based on the evaluations described in this section, the impact of TPBARs on limiting core power distributions for SQN is small and is primarily due to the differences in fuel cycle designs. FRA-ANP's NRC-approved codes and methodology were used to evaluate the acceptability of the SQNTPC cores relative to design limits. Peaking augmentation factors were used to represent the effects of increased peaking due to gaps between TPBAR pencils in the evaluation. The impact on peaking margins is small and similar to those described in the TPCTR. Therefore, it is concluded that there are no significant differences in the conclusions of the evaluation of core power distribution analysis and control for SQN relative to the conclusions reached in the TPCTR.

Reactivity Coefficients

The SQN FSAR (Reference 1) provides the applicable ranges of reactivity coefficients used in the plant safety analyses. The TPCTR provides detailed comparisons of nuclear parameters between TPCs and non-TPCs. The general trends observed in the TPCTR for Doppler and moderator coefficients were also observed in the SQNTPCs. With one exception, which is described below, the reactivity coefficients and kinetics parameters for the TPC designs fall within the bounding ranges provided in the FSAR.

The SQNTPC designs fall within the limits and ranges of the kinetics parameters assumed in the safety analysis except the most negative Doppler-Only Power Coefficient (DOPC). The safety analysis assumption of -19.4 pcm/%FP was exceeded. A most negative value of -21.01 pcm/%FP was calculated for the first transition core at HZP conditions. The impact of this condition is not significant, based on the following evaluation.

At zero power and EOC, flux redistribution causes the DOPC to be more negative than the limit. As power increases, the value quickly returns to within the power dependent limits. Accidents starting at full power are analyzed with the full power DOPC. When the core power changes to zero power after trip, the core shutdown margin is covered by the total reactivity deficit in the shutdown margin calculation. Accidents starting at zero power are conservatively analyzed with a least negative DOPC, because a more negative value will result in a lower final power level. Therefore, the SQNTPC specific value (-21.01 pcm/%FP) of the most negative DOPC exceeding the -19.4 pcm/%FP limit near zero power is acceptable.

Conclusions

The differences between FRA-ANP methodology and that described in the TPCTR are small. The evaluation of the SQNTPC designs shows very similar trends to those established in the TPCTR. The most positive DOPC was not exceeded as seen in the TPCTR. However, the most negative DOPC at HZP conditions was exceeded, but with no impact on safety margin.

Control Rod Worths and Shutdown Margin

Preliminary evaluations indicated that the SQNTPC designs would require the relocation of control rods in order to increase the available rod worth for shutdown margin. The relocation of one group was found sufficient. The RCCAs in core locations symmetric to B12 (Shutdown Bank A, Group 1) would be moved to core locations symmetric to E11 prior to irradiation of TPBARs in SQN. This RCCA movement provides adequate available rod worth for shutdown margin. With this modification, the 1.6 %Δk/k requirement was met with adequate margin.

Conclusions

The shutdown margin requirement for the TPCRD was 1.3 %Δk/k and is 1.6 %Δk/k for SQN. Despite this increase in required shutdown margin, the SQNTPC designs have adequate margin following the proposed RCCA relocation.

Ejected Rod

Analysis of the SQNTPCs during an ejected rod event at HFP indicates satisfactory margin. Evaluations of the HZP ejected rod event for the first transition cycle failed to meet the BOC ejected rod worth requirement. Satisfactory results were obtained by increasing the HZP Rod Insertion Limit (RIL) specified in the Core Operating Limits Report for the first transition core by 8 steps. Figure 2.4.3-3a illustrates the current and the proposed RILs for the SQN plant. The proposed RILs are an example of what would be done to support licensing of the first transition SQNTPC. The results of all other safety and nuclear parameter evaluations were acceptable. Although the results of the demonstration SQNTPC designs indicated a need to modify the RIL based on HZP ejected rod worth, the modification may not be required for all SQNTPC reload designs. Therefore, the need to make a RIL modification will be evaluated during each cycle's reload safety evaluation.

Conclusions

The need to make a RIL modification will be evaluated during each cycle's reload licensing analysis. However, the proposed RIL modifications to meet the ejected rod analysis criteria are small and can be accommodated if necessary.

Effects of Extended Shutdown

The effects of extended shutdown were examined in the TPCTR for the equilibrium cycle design. For an extended shutdown near end-of-life, the buildup of ^{3}He through tritium decay can have a significant impact on core reactivity. The TPCTR showed that the ^{3}He buildup after a six-month shutdown could reduce the critical boron concentration at HFP by about 80 ppm upon startup. This buildup also reduces the cycle energy, since the ^{3}He depletes slowly, much like a burnable absorber.

For the SQNTPCs, the reactivity effects of ^3He buildup will be smaller than those of the TPCTR designs because of the smaller number of TPBARs and the harder neutron spectrum in the fuel lattice. Following a 6 month shutdown at approximately 78% of the cycle length, the core-wide reactivity decrease is approximately –62 ppm boron for the SQN 96-feed equilibrium cycle. The reactivity decrease at mid-cycle is approximately –40 ppm boron for the same cycle. The reactivity effect decreases gradually after return to power. If the effects of plutonium and samarium isotopes are included, a reactivity decrease of -100 ppm is observed after a shutdown. The plutonium and samarium quickly return to equilibrium conditions where the reactivity trends associated with ^3He alone will again dominate. The impact of reduced boron concentrations on most nuclear parameters is beneficial in terms of safety analyses. However, the reactivity effects of an extended shutdown will be evaluated for each reload cycle in the cycle-specific reload safety evaluation.

The power distribution impact of the ^3He buildup is also expected to be small. The effects of ^3He buildup on core power distribution following an extended shutdown were evaluated using the SQN 96-feed maximum TPBAR equilibrium cycle model. Many extended shutdown scenarios would result in a negligible impact on peaking margins. The worst case extended shutdown was found to be six months occurring at approximately 80% of the licensed fuel cycle length. The impact on peaking margins for the worst case was found to be on the order of 2% to 3.5%. Although small, this magnitude is significant enough to require reevaluation of the core power distribution prior to resumption of power operation. Therefore, SQN production TPC designs will be evaluated on a cycle-specific basis relative to the effects of ^3He buildup for extended shutdown. Guidance will be provided on the identification of conditions that could result in the need to reassess core power distribution limits and operational data prior to resumption of full power operation due to ^3He buildup and redistribution following an extended shutdown.

Analyses and testing of irradiated absorber pellets and getters by PNNL show that for core physics calculations, ^3He generated by tritium decay in TPBAR components during a lengthy reactor outage can be assumed to remain in the solid components that contained the parent tritium. During reactor startup and subsequent operation, these TPBAR components (pellets and getters) will begin to release ^3He to the TPBAR free volume, but complete release occurs over a period of days to weeks.

Conclusions

The differences in results between the SQN TPCs and those described in the TPCTR are small and due to the differences in fuel cycle design. The reactivity consequences of ^3He buildup and redistribution after shutdown are dependent on the feed batch size, the number of TPBARs, the ^6Li enrichment used, cycle length, and time in cycle. For reload fuel cycles, guidelines will be provided to specify the conditions under which the core power distribution limits and operational data may require evaluation prior to resumption of full power operation due to ^3He buildup and redistribution following an extended unit shutdown. If an extended shutdown occurs, core operational data and limits will be updated as necessary to ensure that the core is operated within safety analysis and Technical Specification limits.

Summary

In this section, the nuclear design aspects of Sequoyah Nuclear Plant Tritium Production Cores have been presented. The design bases employed are the same as those for current Sequoyah core designs. In the TPC designs, the TPBARs function in a manner that is similar to conventional burnable absorbers. While the depletion behavior of the TPBARs is different than that of conventional burnable absorbers, this does not lead to significant differences in core physics behavior. The behavior of the designs with respect to power distributions, reactivity coefficients, and other core physics parameters is comparable to that of current Sequoyah core designs. Calculation and analysis of key safety parameters have demonstrated that, with the exceptions of shutdown margin, most negative Doppler-Only Power Coefficient (DOPC) at zero power, HZP rod ejection at BOC, and post-LOCA recriticality, the key safety parameters fall within the ranges and limits normally assumed. To ensure that shutdown margin will be adequate, four RCCAs currently located in symmetric peripheral core locations will be moved to the interior of the core so that available inserted rod worth will be greater. Evaluation of the most negative DOPC resulted in no adverse impact on safety limits or margins. The rod ejection evaluation resulted in a small modification to the control bank insertion limits. The post-LOCA recriticality concern was addressed by increasing the minimum RWST boron concentration and cold leg accumulator boron concentration. Therefore, these exceptions do not invalidate the conclusions of the safety analysis. The effects of ^{3}He buildup and redistribution due to extended shutdowns were evaluated and it was concluded that although these effects are small, guidance will be provided to identify the conditions that could result in the need for a reassessment of shutdown margin, power distribution limits and operational data in the event of an extended shutdown. Core limits and operational data would be revised as necessary in the event of an extended shutdown to ensure that core operation remains bounded by the safety analysis and Technical Specification requirements.

Based on these results, it is concluded that viable TPC designs can be developed for Sequoyah that achieve typical cycle energy goals, generate large amounts of tritium, and meet typical design and safety limits.

References

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4. NEMO – Nodal Expansion Method Optimized, BAW-10180-A, Revision 1, B&W Fuel Company, Lynchburg, Virginia, March 1993.

5. Core Operating Limit Methodology for Westinghouse-Designed PWRs, BAW-10163P-A, B&W Fuel Company, Lynchburg, Virginia, June 1989.

6. Production TPBAR Design Inputs for Sequoyah Units 1 and 2, TTQP-1-118, Revision 6, Pacific Northwest National Laboratory, Richland, Washington, September 2000.

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2.4.4 Thermal And Hydraulic Design Evaluation

Introduction

The core thermal-hydraulic performance of SQN Units 1 and 2 was evaluated with respect to the incorporation of the tritium-producing burnable absorber rods (TPBARs) placed in thimble tubes of the FRA-ANP Mark-BW17 fuel assembly design. Analysis results show that acceptable thermal-hydraulic conditions will exist in the transition and equilibrium fuel cycles for TPBAR implementation.

Acceptance Criteria

The thermal-hydraulic evaluation utilizes the following design criteria to demonstrate acceptable operation with TPBARs.

- the mechanical integrity of the thimble tube is maintained during the life of the fuel with the presence of the TPBAR by demonstrating adequate cooling of the thimble tube to preclude excessive component temperatures and corrosion;
- the core will remain protected from departure from nucleate boiling (DNB) by assurance that there will be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling or transition condition during normal operation or anticipated operational occurrence;
- the core departure from nucleate boiling ratio (DNBR) predictions account for the localized fuel rod power influence associated with the positioning of TPBARs within the thimble tubes; and
- centerline fuel melting will not be permitted for normal operation or anticipated operational occurrences.

Methodology

The methodologies used for evaluating the impact of the TPBARs on the thermal-hydraulic environment in the fuel assemblies are consistent with the approved methodologies for licensing the Mark-BW17 fuel design at the SQN units. The LYNXT thermal-hydraulic code (Reference 1), routinely applied to SQN reload licensing analyses, was used to predict the local coolant and surface temperature conditions within the thimble tubes and surrounding subchannels. The BWCMV-A CHF correlation (Reference 2) was also applied in the analysis of the DNBR impact of localized fuel rod power perturbations associated with the

TPBARs using LYNXT. The BWU CHF correlation (Reference 3), approved for application with the Mark-BW17 fuel design, was used for predicting the minimum DNBR for the steamline break (SLB) analysis for the first transition and equilibrium fuel cycles due to its better performance at the low pressure conditions. All remaining DNB analyses utilized the BWCMV-A CHF correlation.

The TACO (Reference 4) and GDTACO (Reference 5) fuel thermal performance codes were used to quantify the impact of TPBAR fuel cycle design steady-state peaking changes on centerline fuel melt limits as compared to non-TPBAR fuel cycles for UO_2 and gadolinia fuel rods, respectively.

In the evaluation of the local coolant and surface temperature conditions within the thimble tubes occupied by TPBARs, a 24-channel LYNXT model was developed that used the conducting-wall feature of the code. The variable-scaled model included a channel representing the thimble tube interior region, 21 individual subchannels around the thimble tube, a channel representing the remainder of the limiting power fuel assembly, and a final channel representing the remainder of the core. Using boundary conditions of a uniform exit pressure and specified core inlet conditions as well as the allowance for lateral crossflow, LYNXT predicted channel flow rates as a function of axial position. This model permitted heat transfer through the thimble tube wall between the channel within the thimble tube and the surrounding four subchannels adjacent to the thimble tube. Coolant exchange was permitted to occur between the interior of the thimble tube and the surrounding subchannels through the thimble tube side holes above the dashpot region. Conservative analysis assumptions included the use of a minimum flow geometry and design peaking in the fuel rods adjacent to the thimble tube occupied by the TPBAR. An axial flux shape sensitivity study was also performed to adequately bound the thimble tube flow rate dependence. Once the axial coolant conditions were established within the thimble tube, TPBAR surface temperatures were determined.

The impact of TPBARs on the magnitude of core bypass flow rate was evaluated to verify that the existing core bypass flow rate assumption used in reload licensing analyses remained bounding and conservative. LYNXT minimum DNBR predictions were also obtained for determining the impact of peaking spikes associated with the axial gaps between the TPBAR pencils on local DNBR. The minimum DNBR sensitivity to the spikes was quantified for a broad range of axial power shapes so that augmentation factors, accommodating the DNBR impact, could be applied in the reload licensing analysis as discussed in Section 2.4.3.

The impact of the presence of TPBARs on centerline fuel melt was examined for UO_2 and gadolinia fuel rods by incorporating the appropriate steady-state radial and axial power peaking for the TPBAR fuel cycle designs into TACO3 and GDTACO fuel rod models used for reload licensing analyses.

The LYNXT code was also used to quantify the magnitude of the steaming rate for SQNTPC and SQNREF fuel cycles to determine whether the TPBAR fuel cycles could be more susceptible to the axial offset anomaly (AOA) phenomenon. The analysis included the relative comparison of SQNTPC fuel cycles with earlier SQNREF fuel cycles.

Results

Analyses show that no bulk boiling will occur in the thimble tube, thereby precluding excessive thimble tube temperatures that could jeopardize the integrity of the tube. The core bypass flow rate through a thimble tube occupied by a TPBAR is comparable to a tube occupied by a thimble plug with little impact on the overall core bypass flow rate. During reload licensing, the cycle-specific core bypass flow rate will be compared to the core bypass flow rate assumption in the DNB analysis of record to assure the analysis of record remains bounding and applicable. The SQNTPC fuel cycles are predicted to be no more susceptible to incur AOA than earlier SQNREF fuel cycles based on steaming rate calculations and the projected boron concentrations.

The magnitude of the augmentation factors attributed to the axial peaking spikes formed by axial gaps between the pencils is generally small and will be applied to fuel rod peaking margin calculations during the reload safety evaluation of SQNTPCs. The evaluation of the TPBAR transition and equilibrium fuel cycles shows acceptable DNBR performance for steady-state and transient conditions.

The centerline fuel melt limits previously established for SQN reloads can be justified for cycles containing TPBARs, therefore, centerline fuel melt limit protection will be assured without additional limitations or constraints relative to existing SQNREF fuel cycles.

Conclusions

FRA-ANP used its NRC-approved codes and methods to compute thimble tube coolant conditions and to demonstrate compliance with the design criteria. Acceptable core thermal-hydraulic conditions are predicted for the operation of TPBARs in future SQNTPCs by the demonstration that all applicable design criteria associated with coolability are met when complemented by a plant-specific/cycle-specific reload licensing evaluation to assure parameter assumptions in the generic analyses remain bounding for the cycles with TPBARs. These include fuel rod integrity, thimble tube integrity, maximum core bypass flow rates, and DNB criteria. The presence of TPBARs in the reload core design did not challenge the DNB criterion. An explicit check of the DNB criterion is included in the cycle-specific reload safety evaluation performed for each SQN reload core. Continued performance of this check will validate the acceptability of each reload core for operation within the DNB design limits.

FRA-ANP did not evaluate the rod withdrawal accident as performed by Westinghouse and discussed in the SER of the TPCTR for demonstrating acceptable DNBR performance. The limiting DNB transient for SQN reload licensing analyses will be examined by FRA-ANP on a cycle-specific basis. FRA-ANP's evaluation did, however, quantify the local and global peaking impact of TPBAR transition and equilibrium fuel cycles.

Although cycle-specific evaluation results are not identified in the TPCTR and SER, FRA-ANP did perform needed analyses to aid in the later cycle-specific analyses. These included the determination of augmentation factors to account for the localized DNB impact associated with the TPBAR pencil axial gaps, the confirmation of acceptable centerline fuel melt limits with TPBAR core configurations, and the assessment of the susceptibility of the fuel cycles to AOA.

References

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4. TACO3 Fuel Pin Thermal Analysis Code, BAW-10162P-A, B&W Fuel Company, Lynchburg, Virginia, October 1989.
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2.9 AUXILIARY SYSTEMS

2.9.1.1 Overhead Load Handling System

The 125/10 Ton Auxiliary Building Crane is the only overhead handling system involved in TPBAR related handling. It handles new fuel assemblies equipped with TPBARs, empty consolidation canisters, the consolidation frame during assembly/disassembly/transport, and shipping casks. The handling of new fuel assemblies and empty consolidation canisters are well within the capacity and are consistent with existing handling procedures for the crane, and therefore require no further evaluation.

Handling of the Consolidation frame in the Auxiliary Building is accomplished within the NUREG-0612 program requirements as embodied in the response to Generic Letter 81-07. Additionally, because handling of the consolidation frame in the cask loading pit is in close proximity to irradiated fuel in the spent fuel pool, additional design considerations/requirements are established as follows:

- The consolidation frame weighs less than ½ of the crane hook capacity. Together with other installed crane safety features, this crane is considered to be equivalent single-failure-proof for this load.
- The lifting device for the consolidation frame will be designed, fabricated, tested, and examined in accordance with ANSI N14.6 for critical loads. The lifting device is considered equivalent single-failure-proof for this lift.

Shipping cask handling considerations are addressed in section 1.5.1.

2.9.1.2 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean up and degasification, reactor coolant make-up, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through a letdown orifice.

The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low-pressure water leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low-pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank (VCT).

In the assessment of CVCS operation at the revised required boron concentrations, the current system design was evaluated to determine if the functional operability of the system and its components are maintained for the TPC.

An operational issue was identified concerning the volume of boric acid required to bring the RCS to the required refueling concentration. The RWST boric acid concentration will be increased to a range of 3600 ppm to 3800 ppm. Before the RWST can be used to fill the refueling cavity, the RCS boron concentration should be raised to RWST boron concentration. This requires more boric acid from the boric acid storage system (boric acid tanks). A calculation of the post LOCA sump pH with the higher boron concentrations indicates that the minimum long term sump pH will be reduced, however, it will remain within the current SQN lower limit of 7.5 pH.

From a "systems" perspective, CVCS operation at the revised boron concentration was reviewed and the results presented in the previous subsection. The overall conclusion from this assessment is that the incorporation of TPBARs will not require any system changes for the CVCS to perform its design basis functions.

2.9.6 Process and Post Accident Sampling System Evaluation

TVA has performed an evaluation of the production of tritium using TPBARs in the SQN Plant and determined that no additional sampling points are needed beyond those presently required by plant technical specifications during the normal plant operating and refueling operations with a Tritium Production Core (TPC). Evaluation of potential leaching of chemical contaminants from TPBARs has determined that the effect of these potential chemical contaminant releases into the Reactor Coolant System or the Spent Fuel Pool will not require any changes to SQN's existing sampling frequencies. However, procedures will be revised prior to TPBAR irradiation to require liquid sampling in the spent fuel pool for tritium while moving and storing irradiated TPBARS. While irradiated TPBARs are stored in the spent fuel pool, tritium sampling will be conducted on a weekly basis. When moving irradiated TPBARs, the spent fuel pool will be sampled daily (TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience). Additionally, action levels will be established in plant procedures to require increased sampling of the Reactor Coolant System (RCS) if tritium concentrations greater than the expected range are noted as indicated in Table 2.9.6-1.

2.11 RADIOACTIVE WASTE MANAGEMENT

2.11.2 Source Terms

Reactor Core

Information to be provided later

Reactor Coolant System

Information to be provided later

Tritium

With respect to tritium sources, in a non-TPC, the production of tritium in the RCS is primarily the result of three processes:

- Ternary fission,
- Boron activation, and
- Lithium activation.

A review of Westinghouse Pressurized Water Reactors benchmark tritium data indicates a nominal production/release tritium value of about 870 Ci/y/unit. This nominal value is consistent with the 845 Ci/y unit average tritium effluent total (Table 2.11.3-2) observed over the past four years (1997 - 2000) at WBN and SQN and will be used in the balance of this discussion.

When reviewing station annual tritium effluents, it is important to recognize that , plants such as WBN and SQN operate with a 18-month fuel cycles which tend to generate more tritium early in the core cycle, owing to higher initial boron concentrations and/or burnable poisons and Integral Fuel Burnable Absorber rods that are required for reactivity control. This results in increasing concentration of tritium in the RCS during the first half of the fuel cycle when discharges from the RCS are relatively small since the amount of feed and bleed necessary to reduce the RCS boron concentration is minimal. However, as the boron concentration is reduced and additional feed and bleed of the RCS is necessary to accommodate boron removal, the amount of primary coolant that is removed increases exponentially and the RCS tritium concentrations are reduced over the latter parts of the cycle.

TPBARs are designed and fabricated to retain as much tritium as possible within the TPBAR. Since the TPBAR produced tritium is chemically bonded within the TPBAR, virtually no tritium is available in a form that could permeate through the TPBAR cladding. However, it is assumed that while operating with a TPC, some of the tritium inventory in the TPBARs may permeate the cladding material and be released to the primary coolant. The design goal for this permeation process is less than 1,000 Ci per 1,000 TPBARs per year. Thus a single TPBAR may release more than 1 Ci/year, but the total release for 1,000 TPBARs will be less than 1,000 Ci/year. As the TPC will contain up to 2,256 TPBARs at SQN, the total design basis tritium input from the maximum number of TPBARs is 2,256 Ci/year into the RCS. The design basis sources of tritium for the RCS, on a fuel cycle basis, are summarized in Table 2.11.2-3.

In addition to the maximum design basis TPBAR permeation release, a potential release scenario is the failure of one or more of the TPBARs. It has been assumed that two TPBARs under irradiation would fail and the entire inventory of tritium would be released to the primary coolant. At the end of the operating cycle, the maximum available tritium in a single TPBAR is calculated to be about 11,600 Ci. While, the occurrence of one or two failed TPBARs is considered to be beyond that associated with reasonable design basis considerations, the assumption of two failed TPBARs is documented in Reference 1.

The TPC projected annual tritium RCS source values are summarized in Table 2.11.2-4.

2.11.3 Liquid Waste Management Systems

TVA has performed an evaluation and determined that for normal TPBAR operation (permeation only), TVA will maintain normal RCS feed and bleed operation for boron control throughout the cycle. Primary coolant discharges volumes with a TPC will therefore be comparable with current plant practice. The maximum tritium level in the RCS, as discussed above under Section 2.11.2, is anticipated to be about 9 μCi/g.

Site-specific data collected during recent extended operating cycles (Watts Bar Unit 1 Cycle 3 and Sequoyah Unit 1 Cycle 10) have provided data to estimate the impact from tritium on station radiological conditions. The RCS maximum tritium levels noted during the extended operating cycles were ≈ 2.5 μ Ci/g with a cycle RCS tritium mean of ≈ 1.0 μCi/g. The end of cycle (pre-flood up) RCS tritium values have typically been in the 0.1 - 0.3 μCi/g range for both Watts Bar and Sequoyah Nuclear Plants. The post-flood up tritium values have typically been in the mid 10^{-2} μCi/g range. The extended cycle tritium peak RCS tritium values of ≈ 2.5 μCi/g have resulted in containment peak tritium Derived Air Concentration (DAC)-fractions of <0.15 for both WBN and SQN with a containment average DAC-fraction of about 0.08. It is understood that containment tritium DAC values are a function of the RCS tritium activity, the transfer of tritium from the RCS to the containment atmosphere (leak rate), and the turnover/dilution of the containment atmosphere through periodic and continuous containment venting and purging.

The projected tritium release to the RCS with a TPC containing TPBARs releasing tritium at the design maximum rate will result in about a factor of four increase over the current tritium production rate, that is,

Ratio = (TPC) 3,126 Ci/yr/(Nominal Core) 870 Ci/yr = 3.6.

By extrapolation (Ratio times the RCS maximum tritium levels noted during extended operating cycles) it has been calculated that with no modifications to TVA's current boron-control feed and bleed methodologies, the design basis RCS maximum tritium values will approximate 9 μCi/g with a cycle mean of $\approx$ 3.6 μCi/g. These values would indicate an estimated containment peak tritium DAC-fraction of $\approx$ 0.6 and an average containment tritium DAC-fraction of about 0.3. The design basis estimated containment average tritium DAC-fraction equates to an effective dose rate of about 0.7 mrem/h.

The TVA TPC estimated end of cycle (pre-flood up) RCS tritium values are projected to be in the 0.4 - 1.2 μCi/g range.

For TPBAR abnormal operation, TVA will establish two tritium RCS action levels > 9 μCi/g and >15 μCi/g. The lower action level will require more frequent sampling (once/day) to monitor the RCS tritium levels. In the unlikely event that the higher action level is exceeded, TVA will take further action to minimize the onsite and offsite radiological impacts of abnormal RCS tritium levels. These actions may include but are not limited to; initiating actions to determine cause, more frequent tritium monitoring of RCS as well as other potentially impacted areas such as containment, increased feed and bleed of the RCS to reduce the tritium concentration, and the temporary onsite storage of tritiated liquids to ensure that the discharge concentration limits are met. The actions levels described above will be used in response to what TVA believes to be extremely unlikely abnormal increases of the tritium levels in the RCS. Plant specific procedures will be developed before TPBAR irradiation utilizing these action levels.

Information to be provided later

In addition, TVA has reviewed the current radioactivity monitoring programs for outdoor liquid storage tanks and has verified that the existing programs provide a appropriate level of assurance with a TPC. The current programs ensure that with an uncontrolled release of the tanks' contents the resulting radioactivity would be less that the regulatory limits at the nearest potable water supply or the nearest surface water supply.

Information to be provided later

Information to be provided later

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These data including a comparison to the station's regulatory established radioactive effluent limits are shown in Table 2.11.3-3.

2.11.4 Gaseous Waste Management Systems Evaluation

As concluded in both the TPCTR and NRC SER, the amount of increase in the radioactive gaseous effluents and the associated dose values are insignificant given the normal evaporative losses from the reactor refueling cavity water and the spent fuel pit water as release paths.

Watts Bar specific data collected during the Lead Test Assembly evaluation program yielded tritium airborne activity levels near the spent fuel pool of less than the detection limit of 1×10^{-9} μCi/ml. The spent fuel pool tritium concentration values over the six month test period averaged around 1×10^{-2} μCi/g.

However, as there is a remote possibility of another release path involving a damaged or dropped assembly or irradiated TPBAR, TVA will monitor for airborne tritium in the spent fuel pool area when moving fuel containing irradiated TPBARs or while consolidating irradiated TPBARs. Prior to initial TPBAR irradiation, TVA will modify the Auxiliary Building and Shield Building Exhaust tritium sampling from periodic effluent grab samples to continuous effluent sampling during periods of release.. Plant specific procedures will be developed before TPBAR irradiation addressing these actions. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

In addition, with regard to the waste gas decay tank, TVA will perform sampling for tritium before releases while irradiating TPBARs. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience. Plant specific procedures will be developed before TPBAR irradiation addressing these actions.

2.11.5 Solid Waste Management Systems Evaluation

For normal TPC operations, the additional solid waste associated with TPCs that TVA will need to handle will be the base plates and thimble plugs that remain after consolidation. TVA will consolidate and temporarily store these items on-site. Offsite shipment and ultimate disposal will be in accordance with established agreements between TVA and DOE. The estimated activity inventory associated with these additional irradiated components (Reference 3) (96 base plates and 48 thimble plugs) (when adjusted to reflect measured dose rate from a Base Plate with 24 Thimble Plugs following 113 day decay adjusted to 180 days) is 4,052 curies per cycle (180 day post irradiation decay) or an average of 2,701 curies per year. This increased activity is associated with metal activation products. The estimated disposal volume of this additional solid waste is 50 cubic feet per TPC operating cycle or an average of 33.3 cubic feet per year. This additional volume is an insignificant increase in the SQN annual estimated solid waste (UFSAR), from 43,550 cubic feet per year to 43,616 cubic feet per year.

TVA's current estimate of the TPBAR cycle work scope includes pre-cycle preparation activities, post cycle removal and handling activities, TPBAR consolidation (including equipment setup and disassembly) and shipping activities, and the processing, packaging, and shipping of the irradiated components for an estimated total of 2,500 man-hours in a 1 mrem/hour radiation field. TVA estimates that on a TPC basis, this additional TEDE is about 1.7 rem per year for TPBAR handling and consolidation activities (2.5 rem per TPC cycle). This estimated additional 1.7 rem per year is an increase of 0.6% of the current SQN station dose assessment of 290 rem (UFSAR), an amount that remains bounded by the station dose assessment of record. Given this small additional ManRem increase for TPBAR handling, consolidation, processing, packaging, and shipping activities, the impact of the increased curies associated with the irradiated components is considered insignificant.

For abnormal TPC operation (TPBAR failure – see Sections 2.11.2 and 2.11.3), where increased feed and bleed operation may be used to reduce tritium levels in the RCS, the increased resins that may result from the increased feed and bleed operation will be stored at TVA in suitable containers. Offsite shipment and ultimate disposal will be in accordance with established agreements between TVA and DOE. As discussed in both the TPCTR and NRC SER, the amount of increase associated with abnormal TPC operation is estimated to be an additional 600 Ci and an additional 30 cubic feet. This additional volume is an insignificant increase in the SQN annual estimated solid waste (UFSAR), from 43,550 cubic feet per year to 43,580 cubic feet per year.

2.11.6 Process and Effluent Radiological Monitoring and Sampling Systems

TVA has reviewed its process and effluent monitoring and sampling equipment program and determined that this program requires minor modifications for a Tritium Production Core (TPC). These changes are limited to the modification of the Auxiliary Building and Shield Building Exhaust tritium sampling from periodic effluent grab samples to continuous effluent sampling, and sample frequency enhancements to the existing monitoring programs, as discussed above under Sections 2.9.6, 2.11.3 and 2.11.4. Plant specific procedures will be developed before TPBAR irradiation addressing these actions. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience. No other changes to TVA's current program are warranted.

Tritium Monitoring

In this section, the various techniques used to monitor for tritium in gases (primarily air) and in liquids are discussed.

Air Sampling

For Tritium air sampling the sampled gas (usually air) must be analyzed for tritium content (usually by liquid scintillation counting). The usual technique is to flow the sampled air through either a solid desiccant (molecular sieve, silica gel, or Drierite) or water or glycol bubblers.

Another available technique for sampling tritium oxide in room air is to use a "cold finger" or dehumidifier unit to freeze or condense the tritium oxide out of the air. When using this methodology, to determine the tritium in air concentration, the relative humidity must be known. The typical lower limit of detection for in-station tritium air samples is 2×10^{-10} µCi/ml.

Liquid Monitoring

Liquids will be monitored by liquid scintillation counting. The typical lower limit of detection for in-station tritium liquid samples is 1×10^{-6} µCi/gm.

Liquid Scintillation Counting

Liquid scintillation counting is a convenient, reliable, and practical way of measuring tritium in the liquid phase. The technique consists of dissolving or dispersing the tritiated compound in a liquid scintillation cocktail, and counting the light pulses emitted from the interaction between the tritium betas and the cocktail. The light pulses are counted by a pair of photomultiplier tubes which, when coupled with a discriminator circuit, can effectively distinguish between tritium betas and those from other sources.

TVA's liquid scintillation counters are periodically calibrated with radioactive sources which are traceable to national standards. The counters are checked periodically with standard radioactive sources in accordance with instrument specific calibration and maintenance procedures.

2.11.7 References

1. DOE/EIS – 0288, March 1999, Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor.

2. Sequoyah Nuclear Plant, Updated Final Safety Analysis Report (UFSAR).

3. Pacific Northwest National Laboratory, 1999, Unclassified Bounding Source Term, Radionuclide Concentrations, Decay Heat, and Dose Rates for the Production TPBAR, TTQP-1-111 Rev. 1.

2.12 RADIATION PROTECTION

2.12.2 Radiation Sources Evaluation

Information to be provided later

2.12.3 Radiation Protection Design Features and Dose Assessment Evaluation

Tritium is a radioactive isotope of hydrogen with a half-life of 12.3 years, which undergoes beta decay, with a maximum energy of 18.6 KeV. The average energy is 5.7 KeV. This low energy limits the maximum range of a tritium beta to about 6 millimeters in air and 0.0042 millimeters in soft tissue. Therefore, the primary radiological significance of exposure to tritium is in the form of internal exposure and the only potential hazard comes when personnel are exposed to open processes that have been wetted with tritiated liquids. Therefore, the design features of the plant that deal with contamination and airborne radioactivity control such as drain and ventilation systems are of potential concern. TVA agrees with the findings of both the DOE topical report and NRC SER that there is negligible impact to these systems by a TPC. TVA has concluded there will be minimal impact on estimated annual Total Effective Dose Equivalent (TEDE) values. TVA has evaluated the additional deep-dose equivalent to select station personnel during TPBAR consolidation and the additional committed effective dose equivalent from possible increased tritium airborne activity in containment. TVA estimates on a TPC basis, this additional TEDE, is about 1.7 rem per year for TPBAR handling and consolidation activities (2.5 rem per TPC cycle) and 1.5 rem per year for the additional committed effective dose equivalent from possible increased tritium airborne activity in containment. This possible additional 6.4 rem per year (two TPCs) is an increase of 2.2% of the current station dose assessment of 290.4 rem (Reference 1) and is considered to be bounded by the station dose assessment of record.

The annual radiological exposure estimates in the TPC Topical Report did not consider additional committed effective dose equivalent, as it was assumed that RCS tritium levels would be maintained at non-TPC levels. The TPBAR handling and consolidation activities were estimated in the Topical Report to require 2 individuals working a single twelve hour shift in a 2.5 mrem/hour radiation field. TVA's estimate of the TPBAR cycle work scope includes; the pre-cycle preparation activities, post cycle removal and handling activities, TPBAR consolidation (including equipment setup and disassembly) and shipping activities, and the processing, packaging, and shipping of the irradiated components for an estimated total of 2,500 man-hours in a 1 mrem/hour radiation field.

2.12.4 Operational Radiation Protection Program Evaluation

TVA has evaluated the current program and determined that there will be no major impact due to inclusion of a TPC. The program modifications are adjustments or changes in scope, rather than major program revisions. Additional monitoring instrumentation and sample equipment to allow better assessment of plant tritium airborne activity will be procured. Plant specific procedures addressing these actions will be developed before TPBAR irradiation.

Tritium Internal Dosimetry Program

A tritium internal dosimetry program requires the determination of the presence or absence of tritium through specific monitoring of the facility and individual workers. It includes the analysis and measurement of tritium in bioassay samples, the evaluation of intakes, and the calculation and assignment of doses from those measurements. It involves evaluation of the intake (Derived Air Concentrations (DACs)), supplemented by the evaluation of bioassay data.

TVA has adopted an evaluation level *(EL)* of 50 mrem committed effective dose equivalent from intakes occurring in a year for employees. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience. The derived limit for the amount of radioactive materials taken into the body of an adult worker by inhalation or ingestion in a year is the Annual Limit on Intake (ALI). One stochastic ALI is equivalent to 5,000 mrem. An intake of a single radionuclide equal to 0.01 of the stochastic ALI or a mixture of radionuclides with a value of 0.01 relative to the stochastic ALI values will yield an *EL*. This is equivalent to 20 DAC hours based on stochastic values

TVA's *EL* is conservative with respect to the guidance provided by the Nuclear Regulatory Commission in Regulatory Guide 8.9, U.S. Nuclear Regulatory Commission, *Regulatory Guide 8.9 – Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.* Regulatory guidance sets the evaluation level at 0.02 of the stochastic ALI. This is equivalent to 40 DAC hours based on stochastic values.

Because of differences in physical properties and metabolic processes, each individual's dose resulting from an internal exposure is unique. In other words, the same radionuclide intake to multiple individuals will likely cause different doses to each individual. However, for very small intakes anticipated, the use of reference man physiological data and biokinetic modeling is adequate to estimate Committed Effective Dose Equivalent, demonstrate compliance with regulatory requirements, and to provide assurance of an appropriate level of protection to workers with respect to internal radiation exposure (References 2 and 3).

Tritium Bioassay Program

The TVA tritium bioassay program will follow the guidance of U.S. Nuclear Regulatory Commission, *Regulatory Guide 8.9 – Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay*

Program. Procedures for the bioassay program will be reviewed and upgraded to ensure sufficient assessment of tritium intake before TPBAR irradiation.

Tritium Monitoring

In this section, the various techniques used to monitor for tritium in gases (primarily air), in liquids, and on surfaces are discussed.

Air Monitoring

Portable ionization chamber instruments will be used for measuring water vapor forms of tritium (HTO) in the station. The output is usually given in units of concentration (typically $\mu Ci/m^3$). Such devices require only an electrically polarized ionization chamber, suitable electronics, and a method for moving the gas sample through the chamber-usually a pump. For real-time tritium monitoring, the practical lower limit of sensitivity range is about one $\mu Ci/m^3$ (0.05 Derived Air Concentration). External background radiation, noble gas, or the presence of radon can reduce the sensitivity of the instrument. TVA has tentatively selected SCINTREX Portable Tritium-in-air Monitor Model 309a, or equivalent, as the instrument of choice.

Air Sampling

Tritium air sampling differs from real-time monitoring in that the sampled gas (usually air) must be analyzed for tritium content (usually by liquid scintillation counting). The usual technique is to flow the sampled air through either a solid desiccant (molecular sieve, silica gel, or Drierite) or water or glycol bubblers.

Another available technique for sampling HTO in room air is to use a "cold finger" or dehumidifier unit to freeze or condense the HTO out of the air. When using this methodology, to determine the tritium in air concentration, the relative humidity must be known. The typical lower limit of detection for in-station tritium air samples is 2×10^{-10} $\mu Ci/ml$.

Surface Monitoring

Tritium contamination will be routinely monitored by smears, which are wiped over a surface and then analyzed by liquid scintillation counting. TVA will develop a routine surveillance program that may include smear surveys in laboratories, process areas, and lunchrooms. In most locations within our facility, weekly or monthly routine smear surveys may be sufficient. The frequency will be dictated by operational experience and the potential for contamination. In addition to the routine survey program, special surveys will be made following spills or on potentially tritium contaminated material being transferred to a less controlled area to prevent the spread of contamination from controlled areas. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

Liquid Monitoring

Liquids will be monitored by liquid scintillation counting. The typical lower limit of detection for in-station tritium liquid samples is 1×10^{-6} µCi/gm.

Liquid Scintillation Counting

Liquid scintillation counting is a convenient, reliable, and practical way of measuring tritium in the liquid phase. The technique consists of dissolving or dispersing the tritiated compound in a liquid scintillation cocktail, and counting the light pulses emitted from the interaction between the tritium betas and the cocktail. The light pulses are counted by a pair of photomultiplier tubes which, when coupled with a discriminator circuit, can effectively distinguish between tritium betas and those from other sources.

TVA's liquid scintillation counters are periodically calibrated with radioactive sources which are traceable to national standards. The counters are checked periodically with standard radioactive sources in accordance with instrument specific calibration and maintenance procedures.

2.12.5 Radiological Environmental Monitoring Program

TVA has reviewed the SQN Radiological Environmental Monitoring Program (REMP) to identify any needed changes to implement the Tritium Production Program. The following REMP changes will be made after receiving NRC license amendment approval but prior to irradiation of the first TPBARs. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

- Atmospheric Moisture - Selected atmospheric sampling stations will be modified to include the collection atmospheric moisture. Collection will be performed at least biweekly.
- Surface Water - Perform tritium analysis on samples collected every four weeks (composite sample collected by automatic sampling system) from the downstream and upstream sampling locations.
- Public Water - Perform tritium analysis on samples collected every four weeks (composite sample collected by automatic sampling system) from downstream public water systems.
- Ground Water - Perform tritium analysis on samples collected every four weeks from the site monitoring wells. Add monthly grab sampling at locations for the nearest (within five mile radius) offsite users of ground water as the source of drinking water.

2.12.6 References

1. Sequoyah Nuclear Plant, Updated Final Safety Analysis Report (UFSAR).
2. National Council on Radiation Protection and Measurements, Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition, NCRP Report No. 87, February 1987.

3. International Commission on Radiological Protection (ICRP) Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation ICRP Publication 54. 1987, Oxford: Pergamon.

2.13 CONDUCT OF OPERATIONS

2.13.1.1 Training

The irradiation of TPBARs will require the review, revision, or development of the following programs:

- Handling, consolidating, and shipping TPBARs.
- General employee training to address TPBAR irradiation.
- Onsite staff training on basic TPC core operation.

As programs and procedures are revised or developed, training will be conducted for TVA personnel. Implementation will include identification/completion of additional training to ensure personnel are adequately trained to perform required activities in a safe and efficient manner.

2.13.1.2 Emergency Planning

TVA has reviewed the Radiological Emergency Preparedness Program (REP) to identify any needed changes to implement the Tritium Production Program. TVA will review and modify actions, action levels, as necessary, based on TPC operating experience. The following REP changes will be made:

- **Dose Codes** - Modify TVA dose codes to include tritium component.
- **Tritium Monitoring & Sampling** - Provide real time offsite tritium monitoring (Scintrex Model 309A or equivalent) and grab sampling (MSA Escort ELF Sampling Pump or equivalent) for TVA and State of Tennessee Field Teams.
- **Sample Analysis** - Establish tritium sample collection, analysis, and interpretation protocols.
- **Procedures** - Modify Emergency Action Levels and decision logic and the Emergency Preparedness Implementing Procedures as required.
- **Training** - Conduct appropriate training for TVA and State of Tennessee Emergency Responder personnel.
- **Dosimetry** - Establish bioassay collection, analysis, and interpretation protocols with respect to tritium for TVA and State of Tennessee Emergency Responder personnel.
- **Validation** - Conduct Tabletop Walkthroughs, Field Sampling Training Exercises, and a joint TVA and State of Tennessee Site Exercise to demonstrate proficiency of tritium-related emergency activities.

2.13.1.3 Administrative, Operating and Maintenance Procedures

Programs, processes, procedures, and instructions will be reviewed and revised as necessary to ensure continued safe operation with a TPC. While some level of tritium already exists in Watts Bar due to normal reactor operations, special cautions will be incorporated into existing procedures as necessary to ensure personnel are aware of activities where tritium production may result in increased tritium levels and associated hazards. The existing administrative process for controlling changes, from identification through implementation, including any required training is not affected by the incorporation of TPBARs.

2.13.2 Safeguards and Security Evaluation

Additional security for the TPBARs will be provided for the period from arrival onsite to installation in the core and the reactor head is installed. Additional security will also be implemented when the head is removed until the TPBARs are shipped offsite. No security measures, in excess of those normally in place, are required while the assemblies are being irradiated. DOE will continue to be the cognizant security agency. NRC's security oversight and responsibilities will remain the same as at all other CLWRs. DOE Chicago has reviewed the Physical Security Plan for TPBARs and revisions are in process. Also, walkdowns of the storage area at Watts Bar and Sequoyah were conducted during their visit for familiarization of these areas and processes. The storage areas were found to be acceptable to DOE during their review.

Material control and accountability of TPBARs will be in accordance with Special Nuclear Material Control procedures which cover shipment, storage, and movement of un-irradiated and irradiated TPBARs, and consolidation of irradiated TPBARs. TVA will revise the Special Nuclear Material Control procedures to describe the actions to be taken by TVA to protect and account for TPBARs while on site.

2.14 INITIAL TEST PROGRAM

2.14.2 Initial Test Program

Testing for the impact of irradiation of a quantity of TPBARs will occur during plant startup with such a core. The monitoring will begin with the TPBARs receipt, continue through low power physics testing, power ascension, and for one cycle of plant operation of approximately 18 months. Routine monitoring will be performed of core power distribution, critical boron, levels of tritium in the RCS liquid and plant environs. Existing procedures are adequate to test and monitor the impact of the TPBARs.

Post-irradiation examination of a representative sample of the TPBAR assemblies will be conducted on site after the first and second cycles. Five to ten percent of the TPBAR assemblies will be visually examined for gross anomalies such as loss of structural integrity or malformation. The need for this surveillance activity will be reviewed after the second production cycle. Changes to this surveillance requirement will be made depending on the results of the previous examinations.

At the conclusion of the fuel cycle, a report that summarizes the behavior of the TPBARs in the reactor and the impact on the plant shall be prepared and made available.

2.15 ACCIDENT ANALYSIS

2.15.2 Safety Evaluation for the Non-LOCA Accidents

The non-LOCA safety analysis parameters have been determined for the Sequoyah reload core design using TPBARs. These parameters were compared to the parameters used in the current applicable safety analysis for Sequoyah (The Fuel Handling Accident is discussed separately in Section 2.15.6.6). This evaluation shows:

1. No changes have been identified in the nominal plant operating conditions (power, coolant temperature, pressure and flow rate) assumed in the plant safety analysis in order to accommodate the TPBARs. Therefore, the existing safety analysis calculations for Sequoyah are not affected by any changes in plant parameters as a result of the TPBARs.

2. No changes to the reactor core thermal hydraulic characteristics or power peaking factors, which could affect the core thermal limits (DNBR and overpower), have been identified as a result of the use of TPBARs. Therefore, the plant thermal limit protection system setpoints do not change as a result of the TPBARs.

3. The nuclear design and fuel rod design calculations performed for the TPBAR reload core design have identified no safety analysis parameters outside of the bounds of the current applicable reload safety analysis parameters. Therefore, no change to the existing licensing-basis safety analysis is required as a result of the TPBAR core design at Sequoyah.

4. Due to post-LOCA subcriticality requirements, the Cold Leg Accumulator and Refueling Water Storage Tank (RWST) boron concentrations are being increased to accommodate the use of TPBARs. This change increases the maximum accumulator boron concentration from 2600 to 3700 ppm and the RWST boron concentration from 2700 to 3800 ppm. No Sequoyah non-LOCA event assumes accumulator actuation. For an increase in the maximum RWST boron concentration, only the non-LOCA events that assume ECCS actuation with maximum boron concentration are potentially affected. The only Sequoyah non-LOCA event that assumes a maximum RWST boron concentration is the Spurious Operation of Safety Injection System at Power event (UFSAR Section 15.2.4). The analysis of this event for Sequoyah assumes that boron is injected into the RCS, via the boron injection tank (BIT) at a boron concentration of 20,000 ppm. The BIT has been removed at Sequoyah, but the analysis inputs were not changed and the boron injection assumption conservatively bounds the increase in RWST boron concentration. The inputs to the current non-LOCA licensing analysis, therefore, are unaffected by the proposed increase in cold leg accumulator and RWST required for the Sequoyah TPBAR core design.

5. FSAR Section 15.3.3 analyses demonstrate that fuel misloadings are low probability events, owing to administrative controls regarding fuel pellet loading in a fuel pin, fuel pin loading in an

assembly, and fuel assembly manufacturing. The analyses also confirm that power distribution effects resulting from misloading events will either be (1) readily detected by the in-core moveable detector system or (2) of a sufficiently small magnitude to remain acceptable and within the design peaking limits. Since, as described above, the inputs to this analysis would not be affected by plant design changes associated with the implementation of TPBARs, the conclusions drawn for the above scenarios would be identical for a TPBAR core at Sequoyah. With the addition of TPBARs at Sequoyah, additional scenarios regarding misloading can be envisioned and the effect of a potential TPBAR cluster misloading should be considered.

A confirming check of the key safety analysis parameters used in the Sequoyah UFSAR analyses for the following non-LOCA events resulted in the conclusion that the TPBAR core design has not changed any of these bounding values. Therefore, the Sequoyah safety analysis for each of these non-LOCA events is unaffected by the TPBAR core design, and all of the applicable acceptance criteria continue to be met.

Transients Unaffected by the TPC

UFSAR Section	Transient
15.2.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
15.2.2	Uncontrolled RCCA Bank Withdrawal at Power
15.2.3	RCCA Misalignment
15.2.4	Uncontrolled Boron Dilution
15.2.5	Partial Loss of Forced Reactor Coolant Flow
15.2.6	Startup of an Inactive Reactor Coolant Loop
15.2.7	Loss of External Electrical Load and/or Turbine Trip
15.2.8	Loss of Normal Feedwater
15.2.9	Loss of Offsite Power to the Station Auxiliaries
15.2.10	Excessive Heat Removal Due to Feedwater System Malfunctions
15.2.11	Excessive Load Increase
15.2.12	Accidental Depressurization of the Reactor Coolant System
15.2.13	Accidental Depressurization of the Main Steam System
15.2.14	Spurious Operation of Safety Injection System at Power
15.3.2	Minor Secondary System Pipe Breaks
15.3.3	Inadvertent Loading of a Fuel Assembly into an Improper Position

15.3.4	Complete Loss of Forced Reactor Coolant Flow
15.3.6	Single RCCA Withdrawal at Power
15.3.7	Steam Line Break Coincident with Rod Withdrawal at Power
15.4.2.1	Rupture of a Main Steam Line
15.4.2.2	Major Rupture of a Main Feedwater pipe
15.4.3	Single Reactor Coolant Pump Locked Rotor
15.4.6	RCCA Ejection

Conclusion

The non-LOCA analyses continue to meet the applicable acceptance criteria for the TPBAR core design.

2.15.5 LOCA Evaluations

2.15.5.1 TPBAR Response to Large and Small Break LOCAs

This evaluation was performed to determine the response of the TPBARs to the design basis LOCAs, both large and small breaks. The TPBAR generates minimal heat during a LOCA and is heated primarily by radiation from the fuel rods to the fuel assembly guide thimble and radiation from the thimble across the gap to the TPBAR. Generally, convection of the steam and entrained liquid on the outer thimble surface provides cooling comparable to that experienced by the fuel rods. However, there are instances when the thimble/TPBAR can be heated, rather than cooled, by the fluid in the surrounding channels. The heatup of the TPBAR was modeled in a conservative fashion using assumptions generally selected to maximize the TPBAR thermal response.

The LOCTA_JR code (Reference 1), which was used to calculate the TPBAR temperatures during a LOCA for the TPCTR, was also used in this evaluation. As a result of their review of the Topical Report, the NRC identified the review of the LOCTA_JR code as an Interface Item for any plant specific implementation of a Tritium Production Core. The LOCTA_JR documentation has since been submitted by TVA for NRC review (see Section 1.5.16).

LOCTA_JR uses as boundary conditions the cladding temperature of the surrounding fuel rods and the core steam and entrained liquid convective heat transfer coefficients and temperatures. The boundary conditions are taken from Appendix K LOCA analyses of record (AOR) for SQN Units 1 and 2.

The following modeling assumptions are made due to the component geometry and the pertinent heat transfer mechanisms:

1. Steam flow in the annulus between the TPBAR and the thimble will be minimal due to (1) the low heat generation rate in the TPBAR and resulting low steaming rates in the annulus and (2) the tendency of

TPBAR swelling to block the annulus. Since steam flow in the annulus would tend to reduce the TPBAR temperatures, it is conservatively neglected.

2. Temperature calculations in the thimble and TPBAR can be performed 1-dimensionally at the elevations of high fuel rod temperature since axial conduction effects are negligible.

3. Heat transfer to the outer surface of the thimble will include radiant heat transfer from the fuel rods and convective cooling from the core steam and entrained liquid flows. The fuel rod temperatures and fluid conditions are boundary conditions to the calculations and are obtained from the Appendix K LOCA analyses.

4. Heat transfer in the thimble/TPBAR annulus consists of radiation and conduction through the steam.

5. Zirc/water oxidation will be calculated on the exterior surface of the thimble. In the thimble/ TPBAR annulus, oxidation of the thimble will be neglected due to the lack of significant steam flow.

6. Heat generation in the TPBAR is included in the thermal calculations although the post-LOCA heating rates in the TPBAR are negligible.

7. Due to the high thermal conductivity of gases within the TPBAR and the low heatup rates, radial temperature gradients inside the TPBAR are minimal. The mean heat capacity of the TPBAR is input as the product of layer weighted density and specific heat, and a mean temperature is calculated.

Because of uncertainties that are inherent with the application of the LOCA hot rod heat transfer coefficient (HTC) to the guide thimble, two cases were run for the LBLOCA. The first case is considered to be a reasonable approach, while the second case was performed to quantify an upper bound response of the TPBARs under LBLOCA conditions. In this second case, the base HTC was modified twice through the transient. From approximately 100 to 120 seconds it was increased by about a factor of 8, after which it was set equal to zero for the remainder of the transient. The purpose here was twofold, 1) to show the overall influences on the transient by variances of the HTC and 2) to attempt to maximize thimble temperature throughout the transient to quantify what the upper bound temperature could possibly be under this extreme.

For LBLOCA, the first case resulted in a guide thimble temperature of 1933°F, while the second case resulted in an upper bound, limiting guide thimble temperature of 2127°F. The corresponding peak TPBAR temperatures for these cases are 1882°F and 2109°F, respectively. It should be noted that the burst model for LOCTA_JR was not used in these runs. The TPCTR provides justification of why TPBAR swelling/burst is expected to be less severe than what would be experienced for the hot rod. The rationale behind this conclusion is still considered to be applicable and therefore no further quantification of this effect is necessary.

Like LBLOCA, two cases for SBLOCA were also analyzed. In the upper bound case, a limiting thimble temperature was determined to be 1040°F with a corresponding peak TPBAR temperature of 1034°F.

This case assumed that HTC=0 from the time of core uncovery to the end of the transient. The other case, which assumes a hot rod HTC on the guide thimble, yields a thimble temperature of 854°F. The peak TPBAR temperature in that case is 832°F. Again the burst behavior, (or lack thereof in this case) depicted in the TPCTR is considered to be applicable in this case as well, particularly because calculated thimble/TPBAR temperatures are less than those presented in the TPCTR.

2.15.5.2 Interaction of TPBARs with LBLOCAs

The TPCTR discussion of the effects of TPBARs on LBLOCAs is still applicable. In addition, an evaluation has been performed considering key core design parameters related to LBLOCAs with respect to TPCs. This evaluation indicates that current and future key parameters can be met for TPCs. In order to maintain post-LOCA subcriticality, the boron concentration in the accumulators is being increased to a range of 3500 to 3800 ppm, and the RWST boron concentration is being increased to a range of 3600 to 3800 ppm. The analysis in support of the post-LOCA long term core cooling requirements demonstrates that the core remains subcritical. (See section 2.15.5.4) As such, it is concluded that the proposed minimum concentrations of 3500 ppm for the accumulators and 3600 ppm for the RWST will be acceptable for the SQNTPC design from a LOCA standpoint. There is no increase in the LBLOCA PCT and the ECCS acceptance criteria limit, dictated by 10 CFR 50.46, continues to be met by the LBLOCA analysis. Therefore, the current SQN Large Break LOCA analysis is applicable to the SQNTPC.

2.15.5.3 Interaction of TPBARs with SBLOCAs

The TPCTR discussion of the effects of TPBARs on SBLOCAs is still applicable. In addition, an evaluation has been performed considering key core design parameters related to SBLOCAs with respect to Tritium Production Cores (TPCs). This evaluation indicates that current and future key parameters for SBLOCA can be met for TPCs. There is no increase in the SBLOCA PCT and the current SQN Small Break LOCA analysis is applicable for the SQNTPC.

2.15.5.4 Effects of TBPARs on Post-LOCA Sump Boron Concentration

The containment sump post-LOCA boron concentration was calculated for the SQNTPCs to ensure that sufficient boron exists in the sump to preclude re-criticality when the Safety Injection pumps are switched from the RWST to the sump for cold leg Safety Injection. Critical boron calculations were performed at post-LOCA conditions versus cycle burnup. The criticality calculations accounted for the number of TPBAR failures due to high LOCA temperatures, 50% ^{6}Li absorber loss through leaching, 100% 3He loss from all failed TPBARs from all failed TPBARs. Moreover, because the rupture of the TPBAR cladding can be energetic, it was conservatively assumed that up to twelve inches of LiAlO2 pellets would be lost from the TPBARs as well (See Section 3.8.3.2). The post-LOCA sump boron calculation considers all sources of liquid that may reach the containment sump following a LOCA and their respective boron concentrations. As indicated in Section 2.4.3 and 2.15.5.2, the boron concentration of the RWST was increased to a range of 3600 to 3800 ppm and the boron concentration of the cold leg accumulators was increased to a range of 3500 to 3800 ppm. With these ECCS changes, the post-LOCA sump boron is sufficient to preclude re-criticality when the Safety Injection pumps are switched to the sump for cold leg Safety Injection at all times in life. This evaluation considers the possibility of sump boron dilution at the time of hot-leg switchover. This evaluation ensures long term core cooling as required by 10 CFR 50.46(b)(5).

Conclusions

Post-LOCA sub-criticality has been demonstrated for SQNTPC designs for the most limiting LBLOCA event. The amount of post LOCA sub-criticality margin (≈120 ppm) for the Sequoyah TPC designs is greater than current SQN designs. Assuming conservative failures of TPBARs and various adverse reactivity conditions, sub-criticality and long term cooling requirements for LBLOCA are satisfied.

2.15.5.5 Effect of TPBARs on Switchover to Hot Leg Recirculation

The inputs in Reference 1 have been incorporated in a new analysis of the core boron build-up to determine the time at which the RHR/safety injection pumps must be aligned to the hot leg in order to preclude precipitation of boron in the Sequoyah Units 1 and 2 post-LOCA core. The post-LOCA LTCC analyses presented herein will remain applicable to Units 1 and 2 so long as the boron concentrations and volumes of the sources of boron remain unchanged.

New post-LOCA LTCC analyses performed for Sequoyah Units 1 and 2 indicate that switchover to hot leg injection recirculation mode cooling post-LOCA must occur 5.5 hours after a LOCA in order to preclude precipitation of boron in the core. Note that this includes the SI interruption duration at switchover to hot leg injection recirculation mode cooling.

It is further noted that after 60 minutes, the charging and HHSI pumps, which take their suction from the discharge of the RHR pumps, can provide sufficient flow to maintain core cooling. Therefore, direct injection into the RCS from the RHRs is not required for hot leg recirculation because the HHSI pumps can provide adequate flow to back flush the core for mitigation of boron precipitation.

Conclusions

The calculations show that the switchover to hot leg injection recirculation mode cooling post-LOCA must occur 5.5 hours after a LOCA.

2.15.5.6 References

1. WCAP-15409, Rev 1, "Description of the Westinghouse LOCTA_JR 1-D Heat Conduction Code for LOCA Analysis of Fuel Rods," September, 2000.

2.15.6 Radiological Consequences of Accidents

This section addresses the potential radiological impact of operation for various design basis accidents with the maximum number of TPBARs installed. The radiological consequences of these accidents are affected primarily by the addition of tritium to the accident source terms. To appropriately account for the radiological consequences of the increased tritium in the TPC, TVA has included calculated Total Effective Dose Equivalent (TEDE) and Federal Guidance Report Number 11 (Reference 1) dose conversion values for thyroid in the accident analysis. TPBARs were designed to withstand the rigors associated with category I through IV events, therefore, no TPBAR failures are predicted to occur during the design-basis accidents except for the large break loss of cooling accident (LBLOCA) or the fuel handling accident. It has been determined that operation with a TPC will not result in exceeding established regulatory guidelines.

2.15.6.1 Loss of AC Power

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a variable parameter. This analysis incorporates assumptions of one percent defective fuel, and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. A realistic analysis is also performed.

Conclusions to be provided later

2.15.6.2 Waste Gas Decay Tank Failure

A waste gas decay tank (GDT) is assumed to develop a leak immediately after a reactor shutdown in which the reactor coolant noble gas inventory has been stored in the tank. Activity is released to the outside atmosphere without any credit for filtration.

The noble gas and iodine activity contained in the GDT is assumed to be unchanged from the existing analysis reported in the FSAR. In addition, consideration is included of tritium in the GDT. The amount of tritium is based on the plant operating with two of the TPBARs having defective cladding so that the tritium leaches into the primary coolant.

Conclusions to be provided later

2.15.6.3 Loss of Coolant Accident

The results of the analysis presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a Loss-of-Coolant Accident (LOCA) do not result in doses which exceed the guideline values specified in a 10 CFR 100.

An analysis based on Regulatory Guide 1.4, 1973, was performed. In addition, an evaluation of the dose to control room operators and an evaluation of the offsite dose resulting from the operation of the Post-Accident Sampling Facility are presented.

Control Room Operator Doses

In accordance with General Design Criterion 19, the control room ventilation system and shielding have been designed to limit deep dose equivalent during an accident period to 5 rem. Thyroid dose is limited to 30 rem and beta skin dose should not exceed 30 rem.

The doses to personnel during a postaccident period originate from several different sources. Exposure within the control room may result from airborne radioactive nuclides entering the control room via the ventilation system. In addition, personnel are exposed to direct gamma radiation penetrating the control room walls, floor, and roof from:

1. Radioactivity within the primary containment atmosphere.
2. Radioactivity released from containment which may have entered adjacent structures.
3. Radioactivity released from containment which passes above the control room roof.

Further exposure of control room personnel to radiation may occur during ingress to the control room from exclusion area boundary and during egress from the control room to site boundary.

Conclusions to be provided later

Environmental Consequences Due to the Operation of the Postaccident Sampling Facility

Section 9.5.10 discusses the Postaccident Sampling Facility (PASF) at Sequoyah Nuclear Plant (SQN). The "worst case" offsite doses resulting from the operation of the PASF are calculated in this section. NUREG-0737 recommends the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guide 1.4 (i.e., 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant). For this "worst case" analysis, the primary system remains intact and pressurized; consequently, the noble gases will stay in the reactor coolant and, in addition, there is no dilution by the Emergency Core Cooling System (ECCS) which would occur during a LOCA.

Conclusions to be provided later

Plant Accessibility Post LOCA

The Sequoyah Nuclear Plant was designed so that access is generally not required outside the main control room for 30 days after an accident. Access to areas within the auxiliary building and structures away from the main complex for the performance of specified tasks are examined individually.

Conclusions to be provided later

2.15.6.4 Main Steam Line Failure Outside of Containment

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a variable parameter. This analysis incorporates assumptions of one percent defective fuel, and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. A realistic analysis is also performed.

Conclusions to be provided later

2.15.6.5 Steam Generator Tube Failure

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generators. A conservative analysis of the postulated steam generator tube rupture assumes the loss of offsite power and hence involves the release of steam from the secondary system. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a variable parameter. This analysis incorporates assumptions of one percent defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system.

Conclusions to be provided later

2.15.6.6 Fuel Handling Accidents

A fuel assembly is assumed to be dropped and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel-handling building ventilation system to the plant vent.

It is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. Also, the assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.7 times core average power. The damaged fuel assembly is assumed to be one with 24 TPBARs which are also assumed to be damaged. Although the release of tritium to the water pool is expected to take place relatively slowly, it is conservatively assumed that the tritium release occurs immediately.

Conclusions to be provided later

2.15.6.7 Rod Ejection Accident (Consequences bounded by 2.15.6.3)

The consequences of a postulated rod ejection accident are bounded by the results of the loss-of-coolant accident analysis.

2.15.6.8 Failure of Small Lines Carrying Primary Coolant Outside Containment

The evaluation of the environmental consequences included the offsite and control room operator dose due to ECCS leakage outside containment following a LOCA.

Conclusions to be provided later

2.15.6.9 References

1. Federal Guidance Report No. 11, "Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhalation, Submersion, And Ingestion", EPA-520/1-88-020, U.S. EPA, Washington, DC.

2.17 QUALITY ASSURANCE

2.17.1 Introduction

Chapter 17 of the SRP deals with the Quality Assurance controls applicable during all phases of a facility's life. Section 2.17.2 and 2.17.3 below, describe the Quality Assurance programs that are applicable to aspects of the TPBAR incorporation and use. TPBARs are being incorporated and used during the Operations Phase, therefore, the applicable portion of the SRP is Chapter 17.2.

Tritium Producing Burnable Absorber Rods (TPBARs) are a basic component as defined by 10 CFR 21. The TPBARs are integral parts of the reactivity control system to keep the reactor core in a safe state, and are therefore, safety-related. In compliance with 10 CFR 21; 10 CFR 50.34(b.6ii); and 10 CFR 50, Appendix A Criterion I, TPBARs are designed, manufactured, and used in accordance with a QA program that complies with the requirements of 10 CFR 50, Appendix B.

After TPBAR irradiation, removal from the reactor core, removal from fuel assemblies, and placement into consolidation containers, TVA prepares irradiated TPBARs for transportation. DOE, as the owner of TPBARs, is responsible for transporting the irradiated TPBARs to the Tritium Extraction Facility. As shipper of record, DOE is responsible for furnishing certified transportation packages for TVA's use in preparing the irradiated TPBARs for DOE's shipment. TVA as a package user maintains and implements an NRC-approved Quality Assurance Program complying with 10 CFR Part 71, Subpart H. Section 2.17.4 below describes the Quality Assurance Program applicable to packaging and transportation of radioactive materials.

2.17.2 Quality Assurance During Operations Phase

Activities, associated with incorporating use of TPBARs at SQN, are performed in accordance with TVA's NRC accepted QA Program (TVA-NQA-PLN89A) which complies with SRP 17.1 and 17.2 and the Fuel Vendor's NRC Approved Quality Assurance Program which complies with SRP 17.1. Activities include but are not limited to establishing the technical, functional, and quality requirements applicable to TPBARs; reviewing and accepting TPBAR design; integrating TPBAR use into facility and reactor core designs and plant operation; obtaining and accepting for use TPBARs that comply with specified technical, functional, and quality requirements; providing applicable control processes and equipment for pre and post irradiation TPBAR handling; and establishing and maintaining protection of the health and safety of workers and the public.

Since DOE procures TPBAR related engineering, design, procurement, fabrication, and delivery services, TVA performs acceptance reviews of applicable DOE documents used to obtain TPBARs and related services to ensure that adequate and acceptable requirements are being identified to the suppliers. TVA evaluates the DOE suppliers for acceptance and placement on TVA's acceptable suppliers list (ASL). The Quality Assurance Program requirements applicable to DOE suppliers associated with TPBAR design and manufacturing is described in Section 2.17.3 below.

TVA procures nuclear fuel and related design and engineering services from NRC licensed fuel vendors who have established and are implementing NRC approved Quality Assurance Programs that comply with 10 CFR 50, Appendix B. The current nuclear fuel vendor for SQN is Framatome ANP, which provides items and services in accordance with its latest NRC approved Quality Management System (QMS).

2.17.3 Supplier Quality Assurance For TPBAR Design and Fabrication

DOE furnishes TPBARs to TVA for irradiation. DOE procures design, material and service procurements, fabrication, assembly, and delivery to TVA or TVA's nuclear fuel vendor. As such, TVA contractually requires that DOE impose TVA's specified technical, functional, quality, and regulatory requirements (including 10 CFR 21) applicable to the TPBARs on DOE suppliers. Provisions are also included for flowing down the applicable requirements to sub-suppliers.

The same QA Program basis used for the Lead Test Assembly TPBAR design, fabrication, and delivery is applied to production TPBARs. DOE suppliers are required to establish, submit to TVA for review and acceptance, and implement a Quality Assurance Program that complies with the requirements of 10 CFR 50, Appendix B; complies with the methods of ASME NQA-1-1994 Basic and Supplementary Requirements; and complies with regulatory positions C.1, C.2, and C.3 of USNRC Regulatory Guide 1.28, Revision 3.

Use of ASME NQA-1-1994 Basic and Supplementary Requirements and the regulatory positions of Regulatory Guide 1.28, Rev. 3 for TPBAR design, fabrication, and delivery has been previously accepted by the NRC as documented in the NRC Safety Evaluation associated with the Watts Bar License Amendment No. 8 (NRC Letter dated September 15, 1997) for TPBARs supplied as Lead Test Assemblies (LTA).

DOE TPBAR and related service suppliers are evaluated by TVA and placed on TVA's acceptable suppliers list (ASL) in accordance with TVA's NRC accepted QA Program. TVA has evaluated and placed on the TVA ASL the Pacific Northwest National Laboratory (PNNL) as an acceptable supplier supporting incorporation of TPBARs into TVA nuclear facilities.

The Pacific Northwest National Laboratory (PNNL) is an acceptable supplier of TPBAR design, material and service procurements, fabrication, and related services. PNNL activities are performed in accordance with the requirements of the PNNL Tritium Target Qualification Project (TTQP) Quality Assurance Manual which has been reviewed and accepted by TVA as complying with the requirements of 10 CFR Part 50, Appendix B; the methods of ASME NQA-1-1994 Basic and Supplementary Requirements; and regulatory positions C.1, C.2, and C.3 of USNRC Regulatory Guide 1.28, Revision 3.

DOE has entered into a contract with WesDyne International LLC (WesDyne), a wholly owned subsidiary of the Westinghouse Electric Company LLC operating under a separate Board of Directors, to become an acceptable supplier of TPBAR design, material and service procurements, fabrication, and related services. TVA is presently evaluating WesDyne as a supplier of core components to be used in TVA power plants as part of the tritium production program. Upon successful completion of the evaluation, WesDyne will be placed on the TVA ASL. WesDyne activities are performed in accordance with the requirements of the latest revision of the NRC accepted Westinghouse Electric Company LLC Quality Management System.

2.17.4 Quality Assurance for Packaging and Transportation of Radioactive Material

DOE owns the TPBARs, procures transportation packages and conveyance services, and is the shipper of record. DOE has contracted TVA to prepare irradiated TPBARs for shipment. The TVA activities associated with packaging and transportation of radioactive materials include preparation of irradiated TPBARs for transportation by loading TPBAR consolidation containers into certified transportation packages, loading and securing the transportation packages onto transport vehicles, performing applicable radiation surveys, and preparation of DOE shipping papers. TVA activities are performed in accordance with TVA's NRC-approved Radioactive Material Package Quality Assurance Plan (PQAP), NRC Docket 71-0227, which complies with 10 CFR 71, Subpart H.

In accordance with the NRC approval of TVA's PQAP, NRC Docket 71-0227, activities such as package design, fabrication, assembly, testing, and modification are satisfied by TVA obtaining certifications from packaging suppliers that these activities were conducted in accordance with an NRC-approved Quality Assurance Program.

Since DOE procures radioactive material transportation packages and related services, TVA identified to DOE the technical, functional, and quality requirements applicable to the transportation package supplier. The requirements include compliance with and package certification to 10 CFR 71 including an NRC-approved QA program. In addition, the DOE supplier(s) are required to be evaluated by TVA and on TVA's acceptable suppliers list (ASL). TVA performs acceptance reviews of applicable DOE documents used to obtain radioactive material packaging and related services to ensure adequate and acceptable requirements are identified to the package supplier. TVA evaluates package suppliers in accordance with TVA's NRC approved Radioactive Material Package Quality Assurance Plan.

Table 2.4.3-1

Core Design and Operating Parameters and Selected Design Limits

Parameter	TPCRD	SQNTPC	SQNREF
Number of fuel assemblies	193	193	193
Number of control rods (RCCAs)	53	53	53
Control rod material	Ag-In-Cd	Ag-In-Cd (80/15/5)	Ag-In-Cd (80/15/5)
Core power level (MWt)	3565	3455	3411
Average linear power density (kW/ft)	5.68	5.51	5.44
Nominal core pressure (psia)	2250	2250	2250
HZP moderator temperature (°F)	557.0	547.0	547.0
HFP core average moderator temperature (°F)	589.7	583	583
Fuel Lattice and Assembly Design	17x17 Vantage+	17x17 Mark-BW	17x17 Mark-BW
Fuel Rod OD (in. cold)	0.360	0.374	0.374
Fuel Pellet OD (in. cold)	0.3088	0.3195	0.3195
Cladding and guide tube Material	ZIRLO™	Zr-4	Zr-4
TPBAR ^{6}Li linear loading (gm/in)	0.30	0.029 and 0.032	N/A
Gadolinia loading w/o Gd_2O_3	NA	4 and 8	6 and 8
IFBA ^{10}B linear loading (g/in)	0.030	N/A	N/A
Active fuel height (in. cold)	144	144	144
Target cycle length (MWd/mtU)	21,564	20,074	21,314
Target effective full power days	494	510*	548**
Core loading (mtU)	81.6	87.8	87.7
Design $F_{\Delta H}$ Limit (with uncertainties)	1.65	1.70	1.70
Design F_Q x P Limit (with uncertainties)	2.50	2.50 x K(z)	2.50 x K(z)
Core control strategy	RAOC	FRA-ANP relaxed offset control***	FRA-ANP relaxed offset control***
Technical Specification MTC limit (pcm/°F)	+7.0 to 70% power +0.0 at 100% power	< 0.0	< 0.0
Shutdown margin requirement (%Δρ)	1.30	1.60	1.60
TPBAR maximum production limit (gm)	1.20	1.20	N/A
TPBAR minimum tritium production limit (gm)	0.15	0.15	N/A
Fuel enrichment limit (w/o ^{235}U)	5.0	5.0	5.0

* 10 EFPD are in power coastdown mode.
** 48 EFPD are in power coastdown mode.
*** Described in Reference 7 from Section 2.4.3.

Table 2.4.3-5

SQNTPC Equilibrium Core
Fuel Batch Description

Batch Identifier*	Fuel Type	Number Of Assemblies	Uranium Rod Initial Enrichment* w/o ^{235}U	Number of Gadolinia Rods per Assembly	Gadolinia Loading, w/o Gd_2O_3	Number of TPBAR Clusters @ Number of Rods x ^6Li Loading, gm/in	Number of TPBARs per Batch	Number of Gadolinia Rods per Batch
3A	**Mark-BW**	**84**	**4.95**	**16**	**4**	**56 @ 24 x 0.032** **16 @ 24 x 0.029** **12 @ 20 x 0.029**	**1968**	**1344**
3B	**Mark-BW**	**8**	**4.95**	**12**	**8**	**--**	**--**	**96**
3C	**Mark-BW**	**4**	**4.95**	**20**	**8**	**--**	**--**	**80**
2A	Mark-BW	84	4.95	16	4	12 @ 24 x 0.029	288	1344
2B	Mark-BW	8	4.95	12	8	--	--	96
2C	Mark-BW	4	4.95	20	8	--	--	80
1A2	Mark-BW	1	4.95	16	4	--	--	16

Fresh fuel is shown in bold.

* Batches 3A, 3B, and 3C are feed; batches 2A, 2B, and 2C are once-burned; batch 1A2 is twice-burned.

Table 2.4.3-6

SQNTPC Equilibrium Cycle
Depletion Summary

(all values are best estimate)

Cycle Burnup (MWd/mtU)	Critical Boron (ppm)	F_Q^N	$F_{\Delta H}^N$	F_Z^N	Axial Offset (%)
0.0	1704	1.787	1.464	1.206	-4.36
150.0	1226	1.737	1.453	1.169	-6.83
500.0	1211	1.727	1.451	1.164	-6.09
1000.0	1192	1.709	1.446	1.157	-5.40
2000.0	1183	1.709	1.437	1.165	-5.52
3000.0	1169	1.688	1.426	1.154	-4.68
4000.0	1155	1.702	1.431	1.160	-4.89
5000.0	1141	1.717	1.436	1.166	-5.22
6000.0	1120	1.724	1.443	1.166	-5.38
7000.0	1081	1.738	1.448	1.169	-5.83
8000.0	1035	1.703	1.444	1.154	-5.28
9000.0	973	1.638	1.430	1.126	-3.82
10000.0	902	1.591	1.413	1.104	-2.65
11000.0	821	1.591	1.411	1.088	-1.19
12000.0	734	1.603	1.429	1.089	-0.42
13000.0	641	1.617	1.443	1.090	-0.03
14000.0	545	1.627	1.451	1.094	0.27
15000.0	451	1.627	1.453	1.092	0.30
16000.0	354	1.631	1.451	1.084	-0.05
17000.0	257	1.626	1.444	1.084	-0.26
18000.0	160	1.625	1.436	1.089	-0.71
19000.0	72	1.568	1.422	1.105	1.31
19680.0	10	1.578	1.415	1.089	0.37
19877.0	10	1.554	1.415	1.115	2.14
20074.0	10	1.598	1.415	1.143	4.07

Table 2.4.3-7

Tritium Production for the First Transition and Equilibrium Cycle Core Designs

Parameter	TPCRD	SQNTPC First Transition Cycle	SQNTPC Equilibrium Cycle
Number of TPBARs	3344	1360	2256
Initial ^{6}Li linear loading (gm/In)	0.030	0.029 and 0.032	0.029 and 0.032
Absorber height (in)	128.5	132.0	132.0
Average ^{6}Li fraction remaining	0.558	0.527	0.553
Average grams of tritium produced per TPBAR*	0.839	0.918	0.889
Peak grams of tritium produced per TPBAR*	1.044	1.026	1.009
Total grams of tritium produced	2805	1248	2007

* No uncertainty applied - best estimate value for a single TPBAR.

Table 2.4.3-8

Nuclear Design Parameters

Parameter Description	SQN Recent Cycle	TPCTR Equilibrium Cycle (ref. 3)	SQN TPC Equilibrium Cycle
Reactivity Coefficients			
Moderator Temperature Coefficients (pcm/°F)			
Near BOL, HZP, No Xenon	-2.0	1.3	-3.5
BOL, HFP, Eq. Xenon	-12.4	-9.9	-14.7
EOL, HFP, Eq. Xenon	-32.7	-32.9	-34.1
Boron Coefficients (pcm/ppm)			
BOL, HZP	-6.6	-6.3	-5.4
BOL, HFP	-6.3	-6.0	-5.1
EOL, HZP	-8.0	-7.6	-6.4
EOL, HFP	-7.6	-7.5	-6.1
Doppler-Only Power Coefficients (pcm/% Power)			
BOL, HZP	-15.7	-11.2	-14.9
BOL, HFP	-8.9	-7.5	-8.9
EOL, HZP	-17.6	-10.	-18.3
EOL, HFP	-7.7	-7.5	-7.9
Total Power Coefficients (pcm/% Power)			
BOL, HZP	-20.8	-15.7	-20.8
BOL, HFP	-16.1	-10.9	-17.8
EOL, HZP	-31.7	-29.8	-33.0
EOL, HFP	-28.5	-24.7	-30.4
Doppler Temperature Coefficients (pcm/°F)			
BOL, HZP	-1.6	-1.7	-1.6
BOL, HFP	-1.6	-1.3	-1.5
EOL, HZP	-1.7	-1.9	-1.7
EOL, HFP	-1.7	-1.5	-1.7
HZP Control Rod Worths (pcm)			
Bank D BOL/EOL*	1042/1095	555/591	1268/1130
Bank C BOL/EOL	1005/921	1148/1147	1144/1119
Bank B BOL/EOL	829/1116	860/851	1109/1400
Bank A BOL/EOL	609/578	645/660	630/478
Shutdown Banks BOL/EOL	2335/2961	3559/3497	3972/4121
* BOL with No Xenon, EOL with HFP Eq. Xenon Note: All values best estimate.			

Table 2.4.3-8

Nuclear Design Parameters (Continued)

Parameter Description	SQN Recent Cycle	TPCTR Equilibrium Cycle (ref. 3)	SQN TPC Equilibrium Cycle
HFP Core Average Neutron Fluxes (n/cm^2-sec)			
BOL			
Thermal	3.64E13	3.67E13	3.04E13
Fast	2.99E14	3.17E14	3.07E14
>1 Mev	7.8E13	8.5E13	8.0E13
EOL			
Thermal	4.31E13	4.23E13	3.45E13
Fast	3.13E14	3.28E14	3.19E14
>1 Mev	8.1E13	8.8E13	8.3E13
Thermal Flux < 0.625 ev, Fast Flux > 0.625 ev			
Boron Concentration (ppm)			
HFP, ARO, BOL, No Xenon Critical	1560	1752	1708
HFP, ARO, BOL, Eq. Xenon Critical	1135	1341	1232
HZP, ARO, BOL, No Xenon Critical	1790	1942	2001
HZP, ARI, BOL, No Xenon k_{eff} = 0.99	1079	1003	681
CZP, ARI, BOL, No Xenon k_{eff} = 0.95	1830	1979^{+}	1905
*50°F, $^{+}$68°F Note: The SQN recent cycle and SQNTPC have difference control rod patterns.			

Note: All values best estimate.

Table 2.4.3-9

Reactivity Coefficients and Kinetics Parameters Values and Ranges Assumed in Reference Plant Transient Analyses

Parameter	Value or Range
Maximum MTC (pcm/°F)	< 0.0 pcm/°F at HZP by Technical Specifications
Most Negative Moderator Temperature Coefficient (pcm/°F)	-45.0
Doppler Temperature Coefficient (pcm/°F)	>-2.2 for LOCA at BOC
Doppler-Only Power Coefficient, (pcm/% power)	
Most Negative	-19.4 to -12.5
Least Negative	-10.2 to -6.5
Delayed Neutron Fraction, β_{eff}	0.0044 to 0.0075

Note: The SQNTPC designs fall within above limits and ranges, with the exception of the most negative Doppler-Only power coefficient at HZP, -19.4 pcm/% power. Section 3.4 addresses this parameter in more detail. An evaluation of the impact of exceeding this limit was performed and found benign.

Table 2.9.6-1

RCS Enhanced Tritium Sampling Program

RCS Tritium Concentration (μCi/g)	Action[*]
Non-TPC	Weekly Sample
TPC < 9 μCi/g [expected range]	Three times a Week
TPC > 9 μCi/g and < 15 μCi/g [upper limit of expected range]	Sample daily
TPC > 15 μCi/g [beyond expected range]	Initiate response to determine causes and activities to mitigate impact. Expand tritium monitoring

[*] Actions and action levels are based on the projected 9 μCi/g maximum tritium concentrations for a TPC. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

Table 2.11.2-1

Comparison of Core Noble Gas and Iodine Activities for a Conventional Core to a Tritium Producing Core

Isotope	Total Core Inventory (Curies)	
	Conventional Core	TPC
Kr 85m		
Kr 85		
Kr 87		
Kr 88		
Xe 133	[*Information to be provided later*]	
Xe 135m		
Xe 135		
Xe 138		
I 131		
I 132		
I 133		
I 134		
I 135		

Table 2.11.2-2

Comparison of Reactor Coolant Noble Gas and Iodine Activities for a Conventional Core to a Tritium Producing Core

Isotope	RCS Activity at Shutdown (μCi/g)	
	Conventional Core	TPC
Kr-85m Kr-85 Kr-87 Kr-88		
Xe-133 Xe-135m Xe-135 Xe-137 Xe-138	*Information to be provided later*	
I-131 I-132 I-133 I-134 I-135		

Table 2.11.2-3

Design Basis Sources of Tritium in the Primary Coolant for the Tritium Production Core Operating Cycle

Tritium Source	Curies
Tritium Producing Burnable Absorber Rods	3,384 (design basis value, actual value will be developed based on operating experience)
Ternary Fission	1,770 (design basis value, actual value is estimated to be 350)
Integral Fuel Burnable Absorbers	40
Control Rods	95
Coolant soluble boron	460
Coolant soluble lithium	176
Deuterium	4
Total Design Basis Tritium	5,929

Table 2.11.2-4

TPC Projected Annual RCS Tritium Source Values

RCS Tritium Sources	Estimated Annual Tritium Release to RCS (Ci)	Estimated Peak RCS Tritium Concentration (μ Ci/g)
Non-TPC with nominal tritium release	870	≈ 2.5
TPC with nominal tritium release and design basis permeation from TPBARs	3,130	≈ 9.0*
TPC with nominal tritium release, design basis permeation from TPBARs and one TPBAR failure having instantaneous release at end of operating cycle	14,730	≈ 53
TPC with nominal tritium release, design basis permeation from TPBARs and two TPBAR failures having instantaneous release at end of operating cycle	26,330	≈ 105

* The projected tritium release to the RCS with a TPC containing TPBARs releasing tritium at the design maximum rate will result in about a factor of four increase over the current tritium production rate, that is, Ratio = (TPC) 3,126 Ci/yr / (Nominal Core) 870 Ci/yr = 3.6.

Table 2.11.3-2

Station Annual Liquid and Gaseous Tritium Effluents (Curies)

SQN	**Liquid**	**Gas**	**Total**	**Gas %**
1997	1559.00	45.29	1604.29	2.82%
1998	1905.00	83.72	1988.72	4.21%
1999	998.00	34.26	1032.26	3.32%
2000	2832.40	62.65	2895.05	2.16%
STATION MEAN	1823.60	56.48	1880.08	3.13%
UNIT MEAN	911.80	28.24	940.04	3.00%
WBN	**Liquid**	**Gas**	**Total**	**Gas %**
1997	639.20	2.56	641.76	0.40%
1998	712.58	7.45	720.03	1.03%
1999	368.43	8.58	377.01	2.28%
2000	1116 00	14.70	1130.70	1.30%
STATION MEAN	694.06	8.32	559.61	1.49%
UNIT MEAN	694.06	8.32	559.61	1.49%
TVA	**Liquid**	**Gas**	**Total**	**Gas %**
PWR UNIT MEAN	839.19	21.61	845.15	2.56%

Table 2.11.3-3

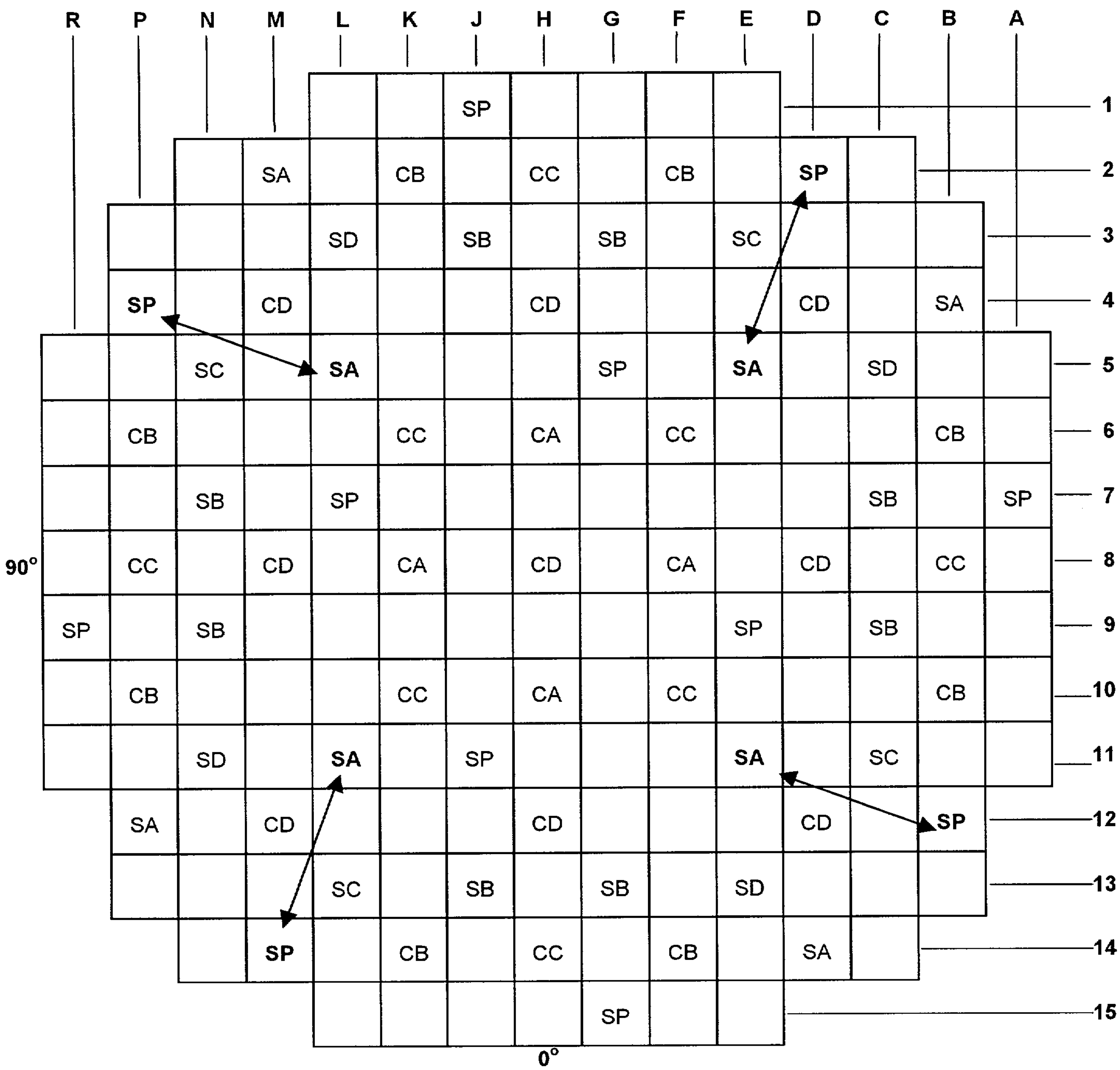
Annual Projected Impact of TPC on Effluent Dose To Maximally Exposed Members of The Public

Pathway - Maximally Exposed Individual	Total Body (mrem)	Critical Organ (mrem)	Annual Regulatory Guidelines (mrem)	Percent of Guideline
Liquid				
Current Core				
TPC				
TPC with one TPBAR Failure				
TPC with two TPBAR Failures				
Current Core (Liver)				
TPC (Liver)	*Information to be provided later*			
TPC with one TPBAR Failure (Liver)				
TPC with two TPBAR Failures(Liver)				
Gaseous				
Current Core (Noble Gases)				
TPC(Noble Gases)				
TPC with one TPBAR Failure (Noble Gases)				
TPC with two TPBAR Failures(Noble Gases)				
Current Core (Bone)				
TPC (Bone)				
TPC with one TPBAR Failure (Bone)				
TPC with two TPBAR Failures(Bone)				

Table 2.15.6-2

Radiological Consequences of a Design Basis LOCA (rem)

	SQN Operation without TPBARs	SQN Operation with 2,256 TPBARs	Acceptance Limit
Site Boundary			
Thyroid dose (ICRP-30)			
– Containment leakage			
– Recirculation leakage			
Total			
Whole body dose (γ)			
– Containment leakage			
– Recirculation leakage			
Total			
TEDE			
Low Population Boundary			
Thyroid dose (ICRP-30)			
– Containment leakage			
– Recirculation leakage			
Total	*Information to be provided later*		
Whole body dose (γ)			
– Containment leakage			
– Recirculation leakage			
Total			
TEDE			
Control Room			
Thyroid dose (ICRP-30)			
– Containment leakage			
– Recirculation leakage			
Total			
Beta-skin			
– Containment leakage			
– Recirculation leakage			
Total			
Whole body dose (γ)			
– Containment leakage			
– Recirculation leakage			
Total			
TEDE			

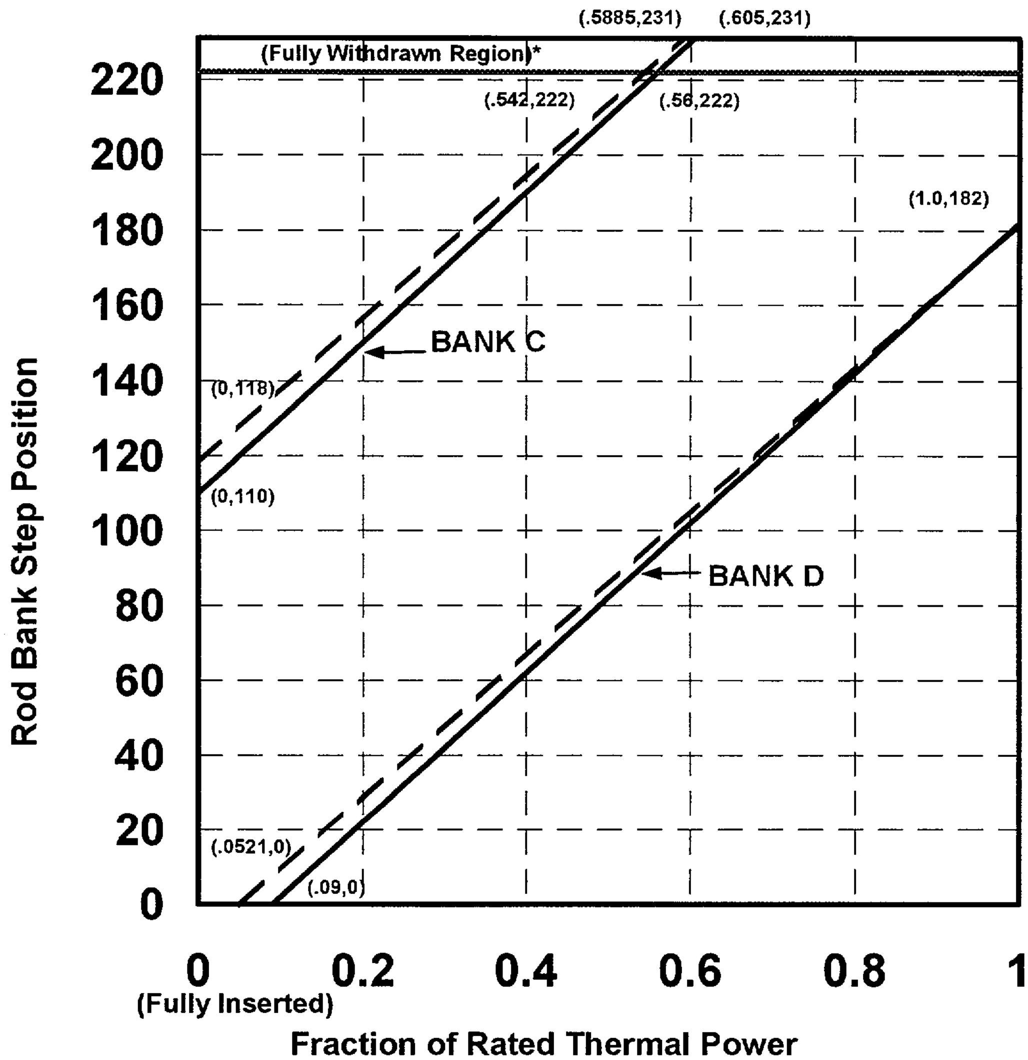


Note: Modified shutdown rod locations are shown in bold.

Bank Identifier	Number of Locations	Bank Identifier	Number of Locations
SA	8	CA	4
SB	8	CB	8
SC	4	CC	8
SD	4	CD	9
		SP (spare)	12

Figure 2.4.3-3

SQNTPC Designs
Control Rod and Shutdown Rod Locations



*Fully withdrawn region shall be the condition where shutdown and control banks are at a position within the interval of ≥222 and ≤231 steps withdrawn, inclusive.

Fully withdrawn shall be the position as defined below,

Cycle Burnup (MWd/mtU)	Steps Withdrawn
≤4000	≥225 to ≤231
>4000 to <14000	≥222 to ≤231
≥14000	≥225 to ≤231

Figure 2.4.3-3a

Rod Bank Insertion Limits Versus Thermal Power, Four Loop Operation

	H	G	F	E	D	C	B	A
8	1A2 16x4%	**3A** **24-.032** **16x4%**	2A 16x4%	**3A** **24-.032** **16x4%**	2A 16x4%	2A 16x4%	**3C** **20x8%**	2C 20x8%
9	**3A** **24-.032** **16x4%**	2A 24-029 16x4%	**3A** **24-.029** **16x4%**	2A 24-.029 16x4%	**3A** **24-.032** **16x4%**	2B 12x8%	**3A** **24-.032** **16x4%**	2A 16x4%
10	2A 16x4%	**3A** **24-.029** **16x4%**	2A 16x4%	**3A** **24-.032** **16x4%**	**3A** **24-.032** **16x4%**	**3A** **24-.032** **16x4%**	**3B** **12x8%**	2A 16x4%
11	**3A** **24-.032** **16x4%**	2A 24-.029 16x4%	**3A** **24-.032** **16x4%**	2A 16x4%	**3A** **24-.032** **16x4%**	2A 16x4%	**3A** **20-.029** **16x4%**	2A 16x4%
12	2A 16x4%	**3A** **24-.032** **16x4%**	**3A** **24-.032** **16x4%**	**3A** **24-.032** **16x4%**	2A 16x4%	**3A** **24-.029** **16x4%**	2A 16x4%	
13	2A 16x4%	2B 12x8%	**3A** **24-.032** **16x4%**	2A 16x4%	**3A** **24-.029** **16x4%**	**3A** **20-.029** **16x4%**	2A BP24x3.5 16x4%	
14	**3C** **20x8%**	**3A** **24-.032** **16x4%**	**3B** **12x8%**	**3A** **20-.029** **16x4%**	2A 16x4%	2A BP24x3.5 16x4%		
15	2C 20x8%	2A 16x4%	2A 16x4%	2A 16x4%				

Notes:

1) Fresh fuel is shown in bold.
2) Batches 3A, 3B, and 3C are feed; batches 2A, 2B, and 2C are once-burned; batch 1A2 is twice-burned.
3) BP24x3.5 indicates 24 Burnable Poison Rods with 3.5 w/o B_4C in Al_2O_3.

Batch ID #TPBARs per assembly ^{6}Li gm/in Fresh #Gad pins x w/o Gd_2O_3

Figure 2.4.3-4

SQNTPC
Equilibrium Cycle Loading Pattern

	H	G	F	E	D	C	B	A
8	1A2 1.082	**3A** 1.129	2A 1.249	**3A** 1.163	2A 1.273	2A 1.294	**3C** 1.150	2C 0.607
9	**3A** 1.129	2A 1.055	**3A** 1.192	2A 1.052	**3A** 1.189	2B 1.218	**3A** 1.048	2A 0.620
10	2A 1.249	**3A** 1.191	2A 1.338	**3A** 1.218	**3A** 1.196	**3A** 1.192	**3B** 1.210	2A 0.605
11	**3A** 1.163	2A 1.051	**3A** 1.217	2A 1.326	**3A** 1.228	2A 1.199	**3A** 0.889	2A 0.422
12	2A 1.273	**3A** 1.189	**3A** 1.196	**3A** 1.228	2A 1.295	**3A** 0.940	2A 0.577	
13	2A 1.294	2B 1.219	**3A** 1.192	2A 1.198	**3A** 0.938	**3A** 0.635	2A 0.244	
14	**3C** 1.150	**3A** 1.048	**3B** 1.209	**3A** 0.888	2A 0.574	2A 0.244		
15	2C 0.607	2A 0.620	2A 0.605	2A 0.422				

Batch ID
Assembly RPD

Batch ID	Number of Assemblies	Power Sharing	Total Burnup	Cycle Burnup
1A2	1	1.082	31,121	0
2A, 2B, 2C	96	1.111	22,795	0
3A, 3B, 3C	96	0.888	0	0

Figure 2.4.3-17

Sequoyah TPC Equilibrium Cycle Assembly Power Distribution at 0 MWd/mtU, HFP, Equilibrium Xenon, Bank CD 215 Steps WD

	H	G	F	E	D	C	B	A
8	1A2 1.064	**3A** 1.103	2A 1.228	**3A** 1.141	2A 1.266	2A 1.302	**3C** 1.161	2C 0.636
9	**3A** 1.103	2A 1.033	**3A** 1.163	2A 1.034	**3A** 1.173	2B 1.225	**3A** 1.058	2A 0.648
10	2A 1.228	**3A** 1.163	2A 1.313	**3A** 1.191	**3A** 1.175	**3A** 1.184	**3B** 1.218	2A 0.632
11	**3A** 1.141	2A 1.033	**3A** 1.190	2A 1.308	**3A** 1.212	2A 1.204	**3A** 0.905	2A 0.447
12	2A 1.266	**3A** 1.173	**3A** 1.174	**3A** 1.212	2A 1.294	**3A** 0.950	2A 0.602	
13	2A 1.302	2B 1.225	**3A** 1.184	2A 1.203	**3A** 0.948	**3A** 0.654	2A 0.260	
14	**3C** 1.161	**3A** 1.058	**3B** 1.217	**3A** 0.904	2A 0.599	2A 0.260		
15	2C 0.636	2A 0.648	2A 0.632	2A 0.446				

Batch ID
Assembly RPD

Batch ID	Number of Assemblies	Power Sharing	Total Burnup	Cycle Burnup
1A2	1	1.063	31,284	163
2A, 2B, 2C	96	0.895	22,928	133
3A, 3B, 3C	96	1.104	167	167

Figure 2.4.3-18

Sequoyah TPC Equilibrium Cycle Assembly Power Distribution at 150 MWd/mtU, HFP, Equilibrium Xenon, Bank CD 215 Steps WD

	H	G	F	E	D	C	B	A
8	1A2 1.019	**3A** 1.142	2A 1.153	**3A** 1.167	2A 1.129	2A 1.117	**3C** 1.132	2C 0.603
9	**3A** 1.142	2A 1.013	**3A** 1.198	2A 1.025	**3A** 1.208	2B 1.115	**3A** 1.059	2A 0.615
10	2A 1.153	**3A** 1.198	2A 1.230	**3A** 1.262	**3A** **1.281**	**3A** 1.242	**3B** 1.192	2A 0.602
11	**3A** 1.167	2A 1.024	**3A** 1.261	2A 1.249	**3A** 1.276	2A 1.155	**3A** 0.945	2A 0.447
12	2A 1.129	**3A** 1.208	**3A** **1.281**	**3A** 1.276	2A 1.221	**3A** 1.014	2A 0.621	
13	2A 1.117	2B 1.115	**3A** 1.241	2A 1.154	**3A** 1.013	**3A** 0.740	2A 0.298	
14	**3C** 1.132	**3A** 1.059	**3B** 1.192	**3A** 0.944	2A 0.619	2A 0.297		
15	2C 0.603	2A 0.615	2A 0.602	2A 0.447				

Batch ID Assembly RPD

Batch ID	Number of Assemblies	Power Sharing	Total Burnup	Cycle Burnup
1A2	1	1.019	40,509	9,388
2A, 2B, 2C	96	0.853	30,664	7,869
3A, 3B, 3C	96	1.147	10,127	10,127

Figure 2.4.3-19

Sequoyah TPC Equilibrium Cycle Assembly Power Distribution at 9,000 MWd/mtU, HFP, Equilibrium Xenon, Bank CD 215 Steps WD

	H	G	F	E	D	C	B	A
8	1A2 0.972	**3A** 1.076	2A 1.053	**3A** 1.092	2A 1.067	2A 1.101	**3C** 1.274	2C 0.720
9	**3A** 1.076	2A 0.949	**3A** 1.103	2A 0.955	**3A** 1.135	2B 1.084	**3A** 1.133	2A 0.720
10	2A 1.053	**3A** 1.102	2A 1.096	**3A** 1.142	**3A** 1.178	**3A** 1.200	**3B** 1.263	2A 0.700
11	**3A** 1.092	2A 0.954	**3A** 1.142	2A 1.115	**3A** 1.185	2A 1.122	**3A** 1.031	2A 0.543
12	2A 1.067	**3A** 1.135	**3A** 1.178	**3A** 1.185	2A 1.172	**3A** 1.078	2A 0.725	
13	2A 1.101	2B 1.084	**3A** 1.200	2A 1.122	**3A** 1.078	**3A** 0.893	2A 0.404	
14	**3C** 1.274	**3A** 1.133	**3B** 1.263	**3A** 1.031	2A 0.724	2A 0.404		
15	2C 0.720	2A 0.720	2A 0.700	2A 0.543				

Batch ID

Assembly RPD

Batch ID	Number of Assemblies	Power Sharing	Total Burnup	Cycle Burnup
1A2	1	0.972	51,316	20,195
2A, 2B, 2C	96	0.866	40,147	17,352
3A, 3B, 3C	96	1.135	22,795	22,795

Figure 2.4.3-20

Sequoyah TPC Equilibrium Cycle Assembly Power Distribution at 20,074 MWd/mtU, 93.4 %FP, Equilibrium Xenon, ARO

SECTION 3 TPBAR EVALUATION

3.1 INTRODUCTION

The TPCTR evaluated the performance of the getter-barrier type TPBARs in a tritium production core loaded with the maximum number of TPBARs possible (~3344). For the tritium production mission in SQN, TVA has determined that the maximum number of TPBARs to be irradiated in the core is 2256. The number of TPBARs to be irradiated in any given fuel cycle will be determined by the core designer, consistent with power plant operations and tritium production requirements.

The differences between the Production TPBAR and the TPC TPBAR described in the TPCTR are:

- Variable pellet stack (pencil) lengths
- Length and material specification for the liner have changed
- Use of a spring clip as an alternative to the plenum spring
- Use of spacer tubes as an alternative to upper and lower getter disks and depleted lithium aluminate pellets
- Reduced the number of pencils in a TPBAR
- Modified top and bottom end plug designs

These changes have been made to improve fabrication processes and to enhance performance. Further details are provided in subsequent sections of this report.

Conclusions

The Production TPBAR design conditions are within the envelope assumed for the TPC TPBAR design conditions given in the TPCTR. The comparison given in Table 1-1 shows that the reactor and core parameters for the TPCRD bound those for SQNREF and SQNTPC. The tritium production, mechanical, and thermal performance design conditions for SQNTPC are within the envelope established in the TPCTR.

Design changes made for the Production TPBARS are a result of TPC TPBAR and Lead Test Assembly (LTA) testing and analyses (see Section 3.10) to improve the ability to fabricate, enhance tritium production, and minimize the potential for non-performance in a production mode.

3.2 PRODUCTION TPBAR DESIGN

3.2.1 Design Description

The TPBAR internal components are a top plenum spacer tube (may also be referred to as a getter tube), a spring clip or a plenum (compression) spring, pellet stack assemblies ("pencils"), and a bottom spacer tube. A pencil consists of a zirconium alloy liner around which are stacked lithium aluminate absorber pellets that are confined in a getter tube, as illustrated in Figure 3.2-1.

Variable Pellet Stack (Pencil) Lengths

The Production TPBAR design uses thin walled annular lithium aluminate ($LiAlO_2$) pellets assembled into stacks, called pencils, extending over the full or partial length of the active core. A single pencil is typically 12 inches in length. The Production TPBAR overall stack lengths of lithium aluminate pellets enriched in ^{6}Li will typically range from 126 to 132 inches.

Length of the Liner and Material Specification

The design length of the production core liner has been tailored for compatibility with the new length dimensions for the absorber pellet stack and getter. The specific dimensions for the length of absorber stack containing ^{6}Li and its offset from the core centerline will be determined by the core designer for compatibility with each future reload core design, therefore small deviations from the dimensions cited in the TPCTR will be required. This flexibility is required to achieve the desired core axial power distribution. The TPCTR specified the liner as "Zircaloy-4." For the production design, the liner is specified as a "zirconium alloy," to provide flexibility in obtaining material. The liner function can be met by any zirconium alloy meeting the specification requirements.

Spring Clip

The use of a spring clip as an alternative to the plenum spring results in more available internal void volume and increases the factors of safety related to internal gas pressure and pressure stresses. The function of both the spring clip and the plenum spring is to provide an axial restraint of the pencil stack during handling and loading operations prior to irradiation. Neither the compression spring nor the spring clip plays a role during or after irradiation.

The spacer tube for the Production TPBAR design is designed to interface with the spring clip or the plenum spring and the top pencil. Dimensions and tolerances on the getters and liners have been changed to facilitate ease of fabrication. All functional requirements relating to dimensional fit-up are satisfied with the revised dimensions and tolerances.

Nickel Plated Zirconium Alloy Spacer Tubes

Depleted lithium aluminate spacers described in TPCTR have been replaced with nickel plated zirconium (NPZ) alloy bottom spacer tubes. A NPZ alloy spacer tube is also used for the top spacer tube in the

Production TPBAR design. These NPZ alloy spacer tubes are preferred structural components and also serve to absorb tritium. Thus, their use allows the option to eliminate the upper and lower getter discs which were used in the LTA for absorbing tritium at the ends of the TPBARs. The NPZ alloy spacer tube occupies less internal void volume than the depleted lithium aluminate spacer. Consequently, the factors of safety related to internal gas pressure and pressure stresses are improved.

Reduced Number of Pellet Stacks (Pencils)

The number of pencils in a TPBAR has been reduced from the description in the TPCTR and in the LTA. The interfaces between the ends of pencils create small gaps in the absorber material. These interface gaps have a minor effect on the power distribution in adjacent fuel rods. Fewer, but longer pencils reduce the number of interfaces between pencils and are preferred to reduce the effect of power peaks in adjacent fuel rods. The number of pencils has been reduced from a total of 12 to 9 standard length and 2 variable length (total of 11) for the first production core. The variable length pencil stacks are positioned so that the pencil-to-pencil gaps occur at different axial locations in three different TPBARs. The TPBARs are arranged on the baseplate in a manner that minimizes power peaking in the fuel rods.

Modified Top and Bottom End Plug Designs

For closure of the TPBARs, end fittings are welded to each end of the cladding tube. The end fittings for the Production TPBARs are manufactured from 316 SS. The top end plug has been modified from the design used in the LTA and the TPCTR designs. The production top end plug design will be compatible with the TPBAR baseplate used by TVA's fuel vendor. The means of attachment of the top end plug to the base plate has been changed from that presented in the TPCTR, and is described in more detail in Section 3.2.3. Additionally, both the top and bottom end plugs are counter bored to increase the internal void volume and decrease mass. The applied stress concentration, vibration fatigue, and flow induced vibration for the modified end plugs satisfy all of the functional requirements for structural integrity.

Future TPBAR Design Enhancements

The thirty-two (32) TPBARs used in the LTA were, for the most part, fabricated and assembled by hand. Such operations would not support the large scale TPBAR production. The changes described above have been made to both improve fabrication and to enhance performance. At the present time, a number of additional enhancements are anticipated for the TPBAR design. These future enhancements are being contemplated for the purpose of improving TPBAR performance, increasing the uniformity of TPBAR quality, lessening the burden of TPBAR irradiation on the host reactor, facilitating the extraction of tritium from TPBARs and improving the capability for large scale TPBAR production.

The future enhancements that are under consideration include the following:

a. Long Getter Tubes

The incorporation of long getter tubes reduces the potential for gaps in the TPBAR absorber which may cause small power peaks in adjacent fuel pins. This design feature removes the need for alternate TPBAR loading patterns and thereby reduces the potential for TPBAR misloading. Advances in fabrication methods will lead to the use of longer pencils, which will improve performance by further reducing the number of pencils and resulting pencil-to-pencil interface gaps in future cores. As fabrication technology matures, steps will be taken to develop full length getters, such that a single pencil will be used, totally eliminating the pencil-to-pencil interfaces.

b. Alternate Plating and Coating Specifications

Alternate plating and coating specifications, which may result in a slightly different product than the current specification, are under consideration as a means to facilitate further improvements in TPBAR performance and provide increased uniformity. The alternate plating and coating specifications offer the potential for increased ease of product inspection, increased margins for mechanical design, and enable TPBAR designs that exhibit enhanced performance. Any alternate plating and coating specification will meet the criteria established for the production TPBARs for chemical compatibility.

c. Alternate Stainless Steel Cladding Materials

The cladding that was used for the LTA, and that which will be used for at least the first production core, is a special order material requiring long lead times to manufacture. For production, the use of more standard cladding material is being investigated, including the use of welded and drawn tubing. Additionally, alternate stainless steel cladding materials offering increased material strength and enhanced corrosion resistance in environments away from the reactor are under consideration as a future TPBAR design enhancement. Enhanced corrosion resistance may provide benefits for those TPBARs exposed to extended moist air storage during transportation or at the tritium extraction facility.

d. End Plug Design Features

A number of changes to the end plug features are anticipated to optimize the fabrication, consolidation, and handling of TPBARs. Refinements to the end plug design will likely be incorporated to facilitate the consolidation of irradiated TPBARs in the spent fuel pool and the handling of the TPBARs in the tritium extraction facility.

Conclusions

Design changes made for the Production TPBARS are a result of TPCTR TPBAR and LTA testing and analyses to improve the ability to fabricate and enhance tritium production. A range of pellet column axial lengths is available for the Production TPBARS to allow core design flexibility and optimization of core power distribution. Mechanical and material changes have been made to the Production TPBAR design to enhance overall performance relative to the TPCTR TPBAR design. The design changes made to the Production TPBAR have been evaluated and determined to meet the functional criteria established by TVA and support the conclusions made by the NRC in the SER related to the TPCTR.

Should TVA, in concert with the TPBAR designer, fabricator, and DOE, conclude that enhancements to the TPBAR design are appropriate, all changes will be evaluated in accordance with TVA procedures.

3.2.2 TPBAR Operation

The irradiation design base case for the Production TPBAR has been increased from 520 effective full power days (EFPD) for the TPCTR design to 550 EFPD. The Production TPBARs are designed to reside in the reactor core for one fuel cycle for a nominal cycle exposure of 510 EFPD, with a maximum exposure of 550 EFPD. For the TPCRD, the expected exposure was 494 EFPD. The capacity factor assumed in the analyses for the TPCRD was 90%. The Production TPBAR has been evaluated assuming a 100% capacity factor for the operating cycle. The extended life-time and exposure limits reflect improvements in the TPBAR design.

Conclusions

The extended life-time and greater capacity factor utilized in the Production TPBAR design reflect more stringent operating conditions than those analyzed in the TPCTR. With these changes, the Production TPBAR design still has adequate margin throughout the operating cycle.

3.2.3 TPBAR Support in the Core Structure

The TPBAR assembly for SQN is shown in Figure 3.2-3. It comprises a maximum of 24 TPBAR rodlets and the upper structure holddown assembly to which the rodlets are attached. For those locations where TPBAR rodlets are not required on a holddown assembly, thimble plug rods are used. The TPBAR assembly design is such that the use of source rods with TPBARs on the same upper structure assembly is precluded. The upper structure assembly is basically the same as that used in the SQN Burnable Poison Rod Assembly (BPRA) to ensure the fuel assembly and SQN reactor mechanical interfaces remain compatible.

The plate portion of the baseplate has 24 tapped holes for attachment of the TPBAR upper end plugs or thimble plugs. The plate is perforated to provide sufficient flow area for the reactor coolant exiting the fuel assembly top nozzle plenum. The flow holes are symmetric with respect to each quadrant of the baseplate and are chamfered at the top and bottom surfaces of the plate to reduce flow turbulence.

The TPBAR upper end plug joint is designed to facilitate harvesting of the TPBAR rodlets. The design consists of the baseplate, crimp sleeve, and threaded stud (upper end plug) as shown in Figure 3.2-4. The baseplate configuration is basically the same as that of the existing Burnable Poison Rod Assembly, with modifications made at the rodlet hole locations. The baseplate thickness is threaded to receive the upper end plug of the TPBAR rod or thimble plug. Crimp sleeves are aligned and welded to the baseplate prior to rod installation. The crimp sleeve consists of an upper thin-wall sleeve and a circular base. The crimp sleeve is welded to the baseplate to prevent removal during the rodlet installation and removal. Therefore the crimp sleeve remains integral to the baseplate during TPBAR harvesting and eliminates additional loose parts. In addition, the baseplate and handling tool interface remain compatible.

Each TPBAR rodlet has an upper end plug that is threaded into and through the baseplate, to which the crimp sleeve is secured. The top portion of the upper end plug is a hex stud to facilitate torqueing and de-torqueing and also serves as the feature to which the sleeve is crimped. The hex stud length is sized for the crimp and torque tool fitups. The upper end plug threads are left-hand such that when the rodlet is removed, conventional right hand torque is used. The threads are designed to minimize the active length and the corresponding stroke used to drive the rodlet out of the baseplate during removal, while ensuring thread structural requirements. Although the thimble plug has a similar design configuration, the length of the hex on the thimble plug terminates just above the crimp sleeve. Therefore, thimble plugs cannot be removed with the TPBAR torque tool and inadvertently mixed with TPBARs during consolidation.

During the consolidation of the TPBAR rods, the rods are detorqued from the baseplate and removed. A hex socket tool is used to de-torque the rodlet using the hex stud on the rodlet upper end plug as the mating feature. Sufficient torque is applied until the resistance of the crimp is exceeded. The rodlet is torqued until it is driven out of the baseplate and into the canister.

If the threaded engagement of the rod to the baseplate becomes galled or is incapable of being removed by conventional methods, a backup method of rod removal is required. To enable rod removal in this case, a small hydraulic cutter would be used to sever the upper end plug of the rod from the baseplate. This method would require that all rods that could be detorqued be removed by the conventional method. Then, the cutter would be delivered onto the rod just below the baseplate. The cutter would sever the upper end plug of the rod at the smallest diameter (a necked down region approximately ½" below the baseplate). Severing the upper end plug in this region would not affect the integrity of the rod itself. This method has been successfully utilized in other spent fuel pool applications. Additional details on TPBAR handling are provided in Section 1.5.1.

Conclusions

The production baseplate differs from both the TPCTR and the current SQN baseplate in the baseplate-to-TPBAR connection design. The TPBAR upper end plug joint is designed to facilitate harvesting of the TPBAR rodlets. This required a modification in the baseplate-to-rod connection as detailed in the above writeup. The connection has been bench tested and verified for interface and functional compatibility.

The changes to the top end plug have been made to simplify the fabrication process and make the TPBARs compatible with baseplate designs of both TVA fuel vendors supporting the DOE Tritium Project. The analyses performed for the TPCRD TPBAR design related to the support of the TPBARs in the core structure are bounding for the Production TPBAR design.

3.3 DESIGN REQUIREMENTS

The Production TPBAR design shall meet the functional requirements listed in Table 3.3-1. These functional requirements are essentially the same as the requirements for the TPC design. The functional requirements for production have been established by TVA. With the exception of functional requirement #6 in Table 3.3-1, it has been confirmed through analyses that all functional requirements are met by the TPBAR design. The completion of an ongoing independent review of the TVA radiological calculations will provide confirmation that functional requirement #6 has also been met. In the TPCTR, permeation through the TPBAR cladding was assumed to be <1.0 Ci/TPBAR/year. For the production design, this nominal release rate is unchanged, but is now presented as "less than 1000 Ci/1000 TPBARs/year." This change reflects the statistical understanding that the release from an individual TPBAR may exceed 1.0 Ci/year, but the total release for 1,000 TPBARs will not exceed 1,000 Ci/year. Table 3.3-2 provides a list of TPBAR design requirements and assumptions for the SQNTPC as well as the TPCRD. Table 3.3-3 compares significant TPBAR parameters for the SQNTPC and the TPCRD.

Conclusions

The production TPBAR design meets the functional requirements established by TVA. Changes in the design requirements reflect the information gained from the LTA fabrication and operational experiences.

3.4 MECHANICAL DESIGN EVALUATION

3.4.1 Tritium Production and Design Life

As noted in Section 3.2.2, the Production TPBAR design life for mechanical evaluation has been changed to 550 EFPD from 520 EFPD used for the TPC design. The nominal design life of the core has been increased to 510 EFPD from the TPC value of 494 EFPD. These changes reflect improvement in the TPBAR design and differences in the operating cycle assumptions between the TPCTR and the plant specific assumptions for the TVA reactors to be used in the tritium production mission.

With a 1.2g tritium/rod limitation, the production TPBAR design evaluations show sufficient design margins up to 550 EFPD.

Conclusions

The Production TPBAR has been evaluated against the plant specific operating parameters for the TVA reactors and will perform with sufficient design margins throughout the operating cycle under all operating conditions.

3.4.3 Absorber Pellets

Evaluation of neutron radiographs for the LTA TPBARs irradiated in Watts Bar confirmed minor cracking of pellets with no evidence of loss of pellet integrity from irradiation and handling. The neutron radiographs also revealed a slight amount of absorber material missing from the top edge of a few pellets in 7 of the 32 irradiated TPBARs. A qualitatively comparable volume of loose absorber material was observed on the bottom getter disk. The maximum volume of loose material in a single TPBAR was estimated to be less than 0.05 cm^3. The loose material is not significant because:

- During irradiation detached lithium aluminate chips are predicted to operate below their melting point.
- Tritium permeation release to the reactor coolant system from pellet material that has relocated to the bottom uncoated end plug is predicted to be negligible.
- The less than 0.05 cm^3 absorber material observed in the bottom of 7 of the 32 irradiated LTA TPBARs is believed to have been abraded from the edge of the top lithium aluminate pellets during fabrication. Implementation of an improved getter end forming process for the production core TPBARs is expected to reduce the potential for these small chips.
- The small amount of material involved will have a negligible impact on core neutronics and power peaks at pencil-to-pencil gaps.

Conclusions

The absorber pellets have demonstrated physical integrity under reactor operating conditions and pre- and post-irradiation shipping and handling. Improvement in the fabrication process is expected to minimize the cracking of the upper pellet surfaces, thus improving performance in the production mission.

3.4.5 Plenum Spring and Spring Clip

The TPCTR design utilized a 302 SS plenum spring to maintain the internals of the TPBAR in place during shipping and handling. This spring is similar in design to those used in BPRA rods and fuel rods. The Production TPBAR has been designed to utilize a zirconium alloy spring clip for the same purpose. The spring clip is also similar to spring clips used in burnable absorber rods. Experimental testing has demonstrated, with high confidence, that the spring clip will provide the restraining force required for pre-irradiation shipping and handling. Neither the plenum spring nor spring clip is required to provide any function during or after irradiation. Sliding of the spring clip along the inner surface of the cladding due to dimensional changes of the pellet stack will not have a negative impact on tritium permeation.

The spring clip occupies less space in the TPBAR than the plenum spring, thus increasing the internal void volume and reducing the internal gas pressure.

Dimensional changes in the plenum spring and spring clip result from thermal expansion and irradiation growth. These phenomena are described in the Materials Properties Handbook (MPH), Reference 1.

Conclusions

The use of a zirconium alloy spring clip in place of the plenum spring reduces the internal gas pressure for the same tritium generation. The spring clip has been designed and tested to provide a restraint to movement of the internal components during pre-irradiation handling and shipping, thus serving the same function as the plenum spring. The spring clip is not required to function during or after irradiation.

3.4.6 References

1. TTQP-7-008, Revision 2, "Material Properties Handbook for the Tritium Target Qualification Project," Pacific Northwest National Laboratory, August 21, 1998.

3.5 TPBAR PERFORMANCE

As described in TPCTR, the TPBARs were designed such that permeation through the cladding would be less than 1.0 Ci/TPBAR/year. For the production design, this value is reported as "less than 1000 Ci/1000 TPBAR/year." While the value of the permeation is not changed from the TPCTR, the new units of reporting emphasize that the release is based on the core average. Thus an individual TPBAR may release more than 1 Ci/year, but the total release for 1000 TPBARs will be less than 1000 Ci/year.

Conclusions

The difference in how permeation from a TPBAR is presented does not impact the total number of curies released. The releases are still bounded by the analyses performed for the TPCTR.

3.5.1 TPBAR Performance Modeling

Hydrogen Ingress from the PWR Coolant

Evaluation of hydrogen (protium) ingress into the TPBARs from the Reactor Coolant System (RCS) as described in the TPCTR assumed that the RCS contained ~35 cm^3 /kg STP of hydrogen. This evaluation for the production design assumes that the RCS contains 50 cm^3 /kg STP of hydrogen. This higher concentration of hydrogen in the RCS provides a higher driving force for hydrogen ingress, and is therefore a more conservative assumption than used in the TPCTR.

Analysis confirms getter loading and internal rod pressure remain within design limits and the preformance of the TPBAR is not adversely affected.

3.5.3 Performance During Abnormal Conditions

During a LBLOCA, those TPBARs which experience conditions of high internal pressure coupled with high cladding temperature will rupture. Burst testing of TPBAR cladding material performed by PNNL conservatively indicates that no more than one pencil worth (~12") of lithium aluminate absorber pellets may be ejected from the TPBAR at the time of the rupture. This loss of pellet material with the leaching of lithium aluminate (at a rate of <3%/day up to 50% of the initial lithium) due to exposure to the RCS coolant has been evaluated and the reactor can still be shutdown and maintained in a safe condition following this event. Further details are provided in Section 2.15.5.4 and 3.7.3.

3.5.4 Failure Limits

Breach of the TPBAR cladding during Conditions I, II, and III is unlikely. However, in the event a TPBAR fails during reactor operation, two TPBAR failure modes have been evaluated to determine the ability to maintain reactor safety. Should a TPBAR fail during operation, it would most likely be due to a small manufacturing or weld defect, which would allow some reactor coolant to enter the TPBAR and TPBAR gases to escape to the coolant. However, there would be no loss of absorber material under these conditions.

In the event of a catastrophic TPBAR failure during reactor operation, all of the lithium is conservatively assumed to be lost immediately to the RCS. Analyses demonstrate the ability to maintain the reactor in a safe condition under both scenarios. See Sections 3.7.3 and 3.8.3.1 for details regarding the effect of pellet leaching on fuel rod performance.

Conclusions

Analyses demonstrate that the reactor can be maintained in a safe shutdown condition even using conservative assumptions related to leaching of lithium and loss of pellet material resulting from TPBAR rupture following a LBLOCA.

3.6 THERMAL-HYDRAULIC EVALUATION OF TPBARS

An evaluation was performed to determine the effects of the representative reactor core thermal hydraulic conditions on the function and integrity of the TPBARs. Approved Framatome ANP analytical tools and methods were applied to calculate the bypass flow through the fuel assembly guide thimble tubes and the thermal performance of the TPBARs located in the guide thimble tubes.

The Framatome ANP methodology was employed to determine for normal operation (Condition I):

- The bypass flow through the fuel assembly guide thimble tubes
- The coolant temperatures in the guide thimble tubes
- TPBAR maximum surface temperatures
- Absence of bulk boiling in the guide thimble coolant flow
- Absence of surface boiling in the guide thimble dashpot

The coolant bulk boiling calculations are performed for the following basic assumptions:

- Thermal core design flow
- Worst-case mechanical TPBAR and guide thimble tubes dimensions and tolerances
- Limiting assembly (containing the hot fuel rod) and the fuel rod power gradient around TPBARs.

Specific evaluation assumptions used in the TPBAR and guide thimble tube evaluation are listed in Table 3.6-2.

Given the conservatism of the input assumptions and parameters discussed above, the evaluation procedure does not require applying additional uncertainties to power, temperature, and pressure which are input at nominal conditions.

Results

TPBARs in the TPC generate higher power than equivalent burnable absorber rods in the same reactor location, primarily due to the higher (n,α) reaction energy release in ^{6}Li than in ^{10}B. Since the external features of both types of rods are almost identical, the guide thimble tube coolant flow remains unchanged. The results of the thermal-hydraulic evaluation are discussed below with respect to the relevant criteria.

No Bulk Boiling

Requirement: There will be no bulk boiling in the guide thimble tubes.

The maximum bulk coolant temperature in the guide thimble tubes is 651.0°F, which is slightly below the saturation temperature of 652.7°F when the TPBAR resides in the limiting fuel assembly containing the hot pin. The maximum cladding surface temperature is 654.4°F.

The TPBAR heat generation (and contribution from the water inside the guide thimble tube) increases the coolant temperature inside the guide thimble. The heat transfer from the adjacent fuel rod channels is a major contributor to the coolant temperature inside the guide thimble.

No Surface Boiling in the Dashpot

Requirement: There will be no surface boiling from the core component rod within the dashpot region of the guide thimble tubes.

The calculated rod surface temperature in the dashpot region of ~ 600°F is well below any surface boiling temperatures.

Bypass Flow

Requirement: The sum of the bypass flow through all the different types of guide thimble tubes, core component rods and the instrumentation tubes in the core shall not exceed the limits specified.

The design basis for the core thermal hydraulic design is a core design bypass flow limit of 7.5% of the reactor flow. The evaluation for the TPBAR transition and equilibrium cores showed that this limit was met with margin.

TPBAR Temperature

Requirement: The maximum temperature of the TPBAR components shall not exceed the melting temperature of component materials during Condition I or Condition II and III events.

Guide thimble inlet and outlet coolant temperatures are used as the boundary conditions with a linear distribution between the top and bottom of the TPBAR. Using this coolant temperature profile and predicted heat inputs from the (n,α) reaction and the gamma heating, rod component temperatures at axial nodes along the TPBAR can be calculated. The nodal component temperatures are then used to predict average gas temperatures at representative burnup steps.

Conclusion

Standard analytical methods used in the nuclear industry were used to evaluate conditions such as bulk boiling during Condition I operation to ensure that an adequate safety margin exists in the thermal-hydraulic design relative to the criteria. These criteria are similar to those that apply to the Framatome ANP BPRAs.

The analyses concluded that the operation with TPBARs in the core is compatible with the TPCTR performance capability and with the current Framatome ANP Mark-BW17 fuel design at the SQN units. The TPBARs meet the functional requirements established by TVA.

3.7 NUCLEAR DESIGN INTERFACES AND CONDITIONS

3.7.1 Lithium-6 Pellet Loading Tolerance Requirement

The ^6Li loading, in grams/inch, of 0.030 for enriched pellets in the TPCTR has been revised to a range of 0.028 to 0.040 ±0.00125. The specific value of the ^6Li loading is determined by the TPBAR tritium production requirements and the core design parameters. The specific value for fabrication is selected based on each core design and is specified by the core designer. For the SQN equilibrium core, the ^6Li loadings are given in Table 3.3-3. The core designer also specifies the axial offset of the TPBAR pellet column.

Conclusions

The change in lithium loading provides needed flexibility to the core designer and does not adversely impact the results of prior safety evaluations. The tritium generated in any individual TPBAR is still limited to 1.2 gm.

3.7.2 Allowable Fuel Peaking Caused by Axial TPBAR Pellet Gaps

As discussed in the TPCTR, axial gaps between absorber pellets in a pellet stack or between pellets in adjacent TPBAR pencils can cause increased local power peaking, called spikes, in adjacent fuel rods. In general, the closer a fuel rod is to a TPBAR location, the larger the potential spike. A given fuel rod may be affected by more than one TPBAR gap, depending on its location in the fuel assembly. If gaps from more than one TPBAR contribute to the local peaking increase in a given fuel rod, a reinforcement of the spike occurs as a consequence of the co-located axial gaps. A functional requirement for the production TPBAR is that "the production design TPBAR shall not cause adjacent fuel to exceed specified acceptable design limits." The application of three TPBAR loading configurations in the production design and the systematic distribution of these three designs within the fuel assembly provide the core designer with flexibility to control the location of pencil-to-pencil gaps and minimize the potential for reinforcement of local peaking due to axially co-located gaps. Analyses performed by the plant fuel vendor ensure that the local peaking factors do not exceed acceptable design limits.

The production design will use fewer pencils in the TPBAR, thus reducing the number of pencil-to-pencil gaps. Ongoing development of the fabrication process is expected to lead to long getters such that only one pencil will be required, thus eliminating pencil-to-pencil gaps.

Conclusions

This change in the loading configuration for TPBARs provides the core designer with flexibility to minimize the impact of pencil-to-pencil gaps on fuel peaking in adjacent fuel rods. This change has a positive impact on plant operation, when compared with the TPC design.

3.7.3 Interfaces and Operational Impacts

TPBAR Failures during Normal Operation

In the event of a catastrophic failure of the TPBAR cladding, recent test data (see Section 3.8.3.2) suggest that significant leaching of lithium from the TPBAR is possible. Accordingly, the safety implications of TPBAR failures with respect to core reactivity and fuel rod integrity were examined. TPBAR failures are extremely unlikely during normal plant operation due to the high reliability of burnable absorber components. Furthermore, in the unlikely event of a TPBAR failure, the following conclusions can be drawn: (1) the implications on global core reactivity are insignificant, and (2) the local power perturbation caused by the catastrophic failure of one TPBAR is sufficiently small such that plant operation can continue without challenging normal operation DNBR limits or compromising fuel rod integrity.

Burnable Absorber Reliability

Burnable absorber components have a long history of reliable use in Westinghouse PWRs. Westinghouse has primarily employed two burnable absorber designs: the Burnable Poison Rod Assembly (BPRA) and the Wet Annular Burnable Absorber (WABA). More than 200,000 burnable absorbers of both types have been irradiated. Prior to 1981, approximately 30,000 BPRAs were irradiated. Of these, only two failures were identified in burnable absorbers that were irradiated for one cycle (Reference 2). Both of these failures occurred early in the history of burnable absorbers and were caused by slumping of the borosilicate glass and swelling of the rod, causing the rod to stick in the assembly. Neither of the failures resulted in cladding failure. (Based on this experience the material specification for the borosilicate glass was changed and no further problems were encountered with burnable absorber performance.) No burnable absorber failures have been reported since Reference 2 was issued in 1981.

The TPBAR design is similar to the BPRA design in that both employ stainless steel cladding. TPBARs will be used in the reactor core in the same manner as BPRAs and WABAs, i.e., they will be attached to base-plates and placed in the fuel assembly guide thimbles, primarily in fresh fuel assemblies. Like conventional burnable absorbers, TPBARs will produce helium that will increase the TPBAR internal pressure in a manner similar to BPRAs and WABAs. TPBAR irradiation, however, will be limited to one operating cycle (BPRAs and WABAs are occasionally used for more than one cycle). PNNL designed the TPBARs using the Westinghouse burnable absorber design documentation as a guide, which resulted in a design that has margins equal to or greater than the Westinghouse commercial burnable absorber rods. In addition, PNNL has placed more stringent quality control requirements on the TPBARs than the requirements placed on the commercial burnable absorbers. The Department of Energy has awarded the contract to fabricate TPBARs to WesDyne International, a subsidiary of Westinghouse Electric Company. The TPBARs will be manufactured at the Westinghouse Columbia Plant under subcontract to WesDyne International, using the same Westinghouse procedures and standards that are currently used to manufacture commercial burnable absorbers, ensuring that the commercial experience will be applied to the TPBARs.

Because of their similar construction, design margins, and operating environments relative to conventional burnable absorbers, TPBAR reliability is expected to at least equal the reliability of BPRAs and WABAs.

Frequency of TPBAR Failures in a Tritium Production Core

The high reliability of the commercial burnable absorbers and the application of that experience to TPBARs yields a very low expected frequency of TPBAR failures in a Tritium Production Core (TPC). Based on the fact that no cladding failures have been observed in the 200,000 burnable absorbers irradiated, a conservative 95% confidence upper limit for the probability of a TPBAR failure has been determined to be 1.5E-05. A typical TPC design will have approximately 2300 TPBARs. For a TPBAR failure to have safety margin implications, the failure must occur at a high power location at a limiting time in core life. Also, for multiple TPBAR failures to produce more severe power peaking than a single failure, the failures must occur in adjacent locations. The frequency of two or more adjacent TPBAR failures is considerably smaller than that for a single failure. The estimate of failure frequency for a single TPBAR in a high power location is 2.9E-03 per year per core, and for multiple adjacent TPBARs in high power locations the estimated failure frequency is 1.2E-07 per year per core. In light of these frequencies, multiple adjacent TPBAR failure scenarios in high power locations are judged to be so improbable that they are not considered credible and further analysis is not warranted. The safety implications of single TPBAR failures are considered below.

Core Reactivity Implications of TPBAR Failures

The global core reactivity effects of a catastrophic TPBAR failure were examined for the TPC designs described in Section 2.4.3. The analyses performed demonstrate that, in terms of global core reactivity, the effect of a TPBAR failure is insignificant. A single TPBAR failure results in a critical boron concentration increase of less than 1 ppm, assuming that all the lithium leaches from the TPBAR. This small reactivity increase is of no consequence with respect to plant operation or shutdown margin and can be easily accommodated by the plant boron system.

DNB Margin Implications of TPBAR Failures

The power distribution effects of a single TPBAR failure were examined for the Tritium Production Core designs. As Section 3.5.4 discusses, TPBAR failures during normal operation will most likely be due to a small manufacturing or weld defects. Such failures will not result in absorber loss, and so the peaking factor increases due to such defects will be negligible. To assess the DNB margin implications of catastrophic failures, the increase in local power peaking was calculated assuming single TPBAR failures at high power locations in the reactor core and at limiting times in the operating cycle. The results of these evaluations show that single TPBAR failures produce peak fuel rod power increases of 4-6%. The effect of the TPBAR failure is localized and limited to a small number of fuel rods in the immediate vicinity of the failed TPBAR. This local power increase assumes that 100% of the lithium leaches from the TPBAR. This is a very conservative assumption.

The 4-6% increase represents the expected change in the assembly hot rod power due to the local power perturbation caused by catastrophic failure of the TPBAR cladding and complete leaching. For the TPC designs discussed in Section 2.4.3, the normal operation $F_{\Delta H}$ limit was not exceeded for a single TPBAR failure. In addition, the DNB safety limits were not exceeded for a single TPBAR failure, assuming the core parameters were within normal operating limits. This was also verified to be true for operation with the core thermal-hydraulic conditions at the extremes of the DNBR-based safety limits. Thus, single TPBAR failures in TPC designs will not cause normal operating limits to be exceeded, nor will DNBR safety limits be exceeded, assuming normal operation. Therefore, fuel rod integrity will be maintained.

Based on the above, the safety implications of TPBAR failures are judged to be sufficiently small such that normal plant operation can continue without challenging DNBR limits or fuel rod integrity.

Operation with Catastrophic TPBAR Failure

In the unlikely occurrence of a catastrophic TPBAR failure except for very early in the cycle, the increased tritium concentration should be noticed during monitoring of the reactor coolant. Should this occur, plant procedures will be in place to specify the appropriate actions to initiate. The procedures will evaluate conditions and determine appropriate actions such that safety limits would not be exceeded in the event of a moderate frequency event. Therefore, power operation could continue without adverse consequences to fuel design limits.

Conclusions

The frequency of TPBAR failures occurring in a Tritium Production Core is small due to the expected high reliability of TPBAR components. In particular, the frequency of experiencing two or more TPBAR failures at limiting core locations is extremely small, so that such scenarios are not considered credible. The safety implications of single TPBAR failures were examined with the following conclusions:

1. the global reactivity increase is very small, less than 1 ppm, and

2. even with the conservative assumption of complete leaching, the local power peaking due to a single TPBAR failure is such that DNBR safety limits will not be challenged assuming normal operation.

Based on the above, the safety implications of TPBAR failures are judged to be sufficiently small such that normal plant operation can continue without challenging DNBR limits or fuel rod integrity. In the unlikely event of a catastrophic TPBAR failure, plant procedures will specify the appropriate actions required to validate the accident analyses results for continued operation and to ensure that fuel failures would be precluded.

TPBAR Compatibility with RCS Chemistry

During normal operation, TPBARS release a minimal amount of tritium to the RCS coolant. As described in the TPCTR, the TPBARs were designed such that permeation through the cladding would be less than 1.0 Ci/TPBAR/year. For the production design, this value is reported as less than 1000 Ci/1000 TPBAR/year. While the value of the nominal release rate is not changed from the TPC topical report, the new units of reporting emphasize that the release is based on the core average. Thus an individual TPBAR may release more than 1 Ci/year, but the total release for 1000 TPBARs will be less than 1000 Ci/year.

Conclusions

This change in the manner in which the permeation is stated does not change the conclusions from TPCTR.

Refueling Operations

The TPBARs will be handled and shipped to the reactor site by methods similar to those applied to burnable absorbers. Prior to shipment to the reactor, the TPBARs are attached to a baseplate, see Figure 3.2-3, and inserted into fuel assemblies at the fuel fabrication facility. Fuel assemblies may be shipped with TPBARs in guide thimble locations in standard shipping containers for fresh fuel, applying standard procedures. Receipt of the TPBAR clusters/fuel assembly combination will follow TVA's standard receiving, unloading and handling procedures for burnable absorber and fuel assemblies. Additionally, TPBARs may also be supplied in fuel skeletons and relocated into the spent fuel pool utilizing existing procedures and equipment.

During refueling operation, with normal refueling and fuel pool temperatures at approximately 110°F, the tritium release from TPBARs is very low, much less than 1 Ci/TPBAR/year and is not considered to affect any evaluations. Defective TPBARs moved to the fuel pool could continue to release the stored tritium at a slow rate into the pool. To quantify the release of tritium from a breached irradiated TPBAR in the spent fuel pool as a result of mishandling, PNNL conducted laboratory tests with irradiated lithium aluminate absorber pellets in both deionized and borated water to simulate spent fuel pool composition. The rate for leaching tritium from irradiated absorber pellets in simulated PWR spent fuel pool water at 24°C and 93°C demonstrated that if a handling accident resulted in simultaneous breaching of 24 TPBARs (one full baseplate) in the spent fuel pool, the tritium concentration in the pool will remain below the 60 μCi/ml TVA action level at all times following the breach. The 60 μCi/ml spent fuel pool tritium activity action level was established to maintain the refueling floor airborne activity below the 10 CFR 20 limit for an airborne radioactivity area.

Conclusions

During refueling operations, TPBAR assemblies will be handled in the same manner as burnable poison assemblies. The analyses performed have evaluated the impacts to the spent fuel pool and surrounding area resulting from damage to 24 TPBARs due to a handling accident. The analysis and the effects will be provided later. See Section 2.15.6.6, "Fuel Handling Accidents."

On-Site TPBAR Assembly Movement and Handling

Handling, consolidating, and preparation for off-site shipment of TPBARs will be controlled in accordance with the plant's procedures (see Section 1.5.1). Weights and interface dimensions of fuel assemblies containing TPBARs are within design parameters of the existing handling equipment and therefore no new or modified tooling or procedures are required for the movement and handling of fuel assemblies with TPBAR clusters. The tooling and procedures required to relocate burnable poison rod assemblies (BPRA) is sufficient to handle TPBAR clusters between fuel assemblies.

Conclusions

On-site TPBAR assembly movement and handling is similar to processes being used at the plant to move BPRAs.

Off-Site Shipping of TPBAR

After removal from the fuel assemblies, TVA will load TPBARs into a consolidation canister, which will be loaded into a shipping cask. Off-site shipment of TPBARs is not a TVA responsibility and will be executed by DOE or an agency assigned by DOE.

One approach for loading and shipping the TPBAR clusters requires a cask outfitted in a manner similar to that used for the LTA shipment. For a larger number of TPBARs, a shipping cask may be manufactured to receive a consolidation canister(s) capable of holding up to 300 TPBARs each. A crane will be used to handle the cask in the facility in accordance with plant procedures and requirements for handling heavy loads in safety related areas.

Conclusions

The process of consolidating TPBARs into a consolidation canister for loading into a shipping cask is a new step and involves new equipment. Analyses have been performed to evaluate the effect of damage to a dropped assembly and a dropped canister.

Information to be provided later

(See Section 1.5.1)

TPBAR Absorber Material Relocation

An evaluation of the neutron radiographs for the LTA TPBARs irradiated in Watts Bar confirmed that there was minor cracking of pellets with no evidence of loss of pellet integrity from irradiation and handling. The neutron radiographs also revealed a slight amount of absorber material missing from the top edge of a few pellets in 7 of the 32 irradiated TPBARs. A qualitatively comparable volume of loose absorber material was observed on the bottom getter disk. The maximum volume of loose material in a single TPBAR was estimated to be less than 0.05 cm^3. As noted in Section 3.4.3, this loose material does not create a neutronics problem, nor does melting of the loose material occur. Further destructive analysis of the pellets will be performed over the next year. No densification or phase changes of the absorber ceramic over the temperature range of the operating conditions was observed from earlier tests and nothing in the observations of the LTA TPBARs to date would indicate that such effects will be found.

Conclusions

Some minor cracking of pellets was observed and a small amount of pellet material was found to have relocated to the bottom of some of the LTA TPBARs. This material is believed to have been abraded from the edge of the top lithium aluminate pellets during fabrication. Implementation of an improved getter end forming process for the production core TPBARs is expected to reduce the potential for these small chips. As noted in Section 3.4.3, the minimal amount of material involved does not create a problem for reactor operations.

Loss of Coolant Events

During a cold leg break, substantial heat-up of the TPBAR cladding is possible. As discussed in Section 3.8.3.2, cladding breach can occur at LOCA conditions if the cladding temperature and internal pressure of the TPBARs reach limiting values. Consequently, post-LOCA critical boron calculations were performed for the Sequoyah TPC equilibrium and transition cycles which conservatively identified TPBAR failures as a function of burnup with resultant leaching of 50% of the contained ^6Li and loss of twelve inches of $LiAlO_2$ pellets. The calculations demonstrated subcritical margin throughout the cycle.

Conservatisms in this analysis included 1) a conservative estimate of the number of failed TPBARs versus burnup, 2) a complete loss of ^3He from all failed TPBARs, 3) a full twelve inches of $LiAlO_2$ absorber ejected from the TPBAR, 4) a conservative reactivity model for a failed TPBAR rodlet, and 5) no credit is taken for control rods. Furthermore, the location of the ejected absorber material is modeled at the most reactive axial location in the core, near the top of the TPBAR absorber column. The most likely failure location is at the pre-LOCA axial peak near the mid-plane of fuel. In addition, it is expected that the control rods will insert for a cold leg break due to the low forces on the reactor upper internals, providing additional sub-critical margin.

For a hot leg break, the control rods may not insert. However, heat-up of the TPBAR cladding is not expected and therefore no TPBAR failures (and subsequent loss of lithium) would occur.

Conclusions

The amount of post LOCA sub-criticality margin (≈120 ppm) for the Sequoyah TPC designs is greater than that for current SQN designs. Identification of conservative assumptions in the analysis supports the expectation that additional post-LOCA subcriticality margin is available. See Section 2.15.5.4 for further discussion of this analysis.

Handling Damage of TPBARs

Calculations performed to support the design of a consolidation container indicate that a TPBAR can survive a drop from a height of ~1.7 feet without significant damage. Calculations also show that a consolidation canister filled with TPBARs (~300) can survive a lateral acceleration limit of 50 g and an axial acceleration of 60 g, thus TPBAR damage will not occur as a result of normal handling and shipping operations.

To quantify the release of tritium from a breached irradiated TPBAR in the spent fuel pool as a result of mishandling, PNNL conducted laboratory tests with irradiated lithium aluminate absorber pellets in both deionized and borated water to simulate spent fuel pool composition. The rate for leaching tritium from irradiated absorber pellets in simulated PWR spent fuel pool water at 24°C and 93°C demonstrated that if a handling accident resulted in simultaneous breaching of 24 TPBARs (one full baseplate) in the spent fuel pool, the tritium concentration in the pool will remain below the 60 μCi/ml TVA action level at all times following the breach. Following such an event, TVA will take the necessary steps to stop the leaching of tritium and return tritium levels in the SFP to normal.

Conclusions to be provided later

3.7.4 References

1. TTQP-1-116, Revision 8, "Production TPBAR Inputs for Core Designers," Pacific Northwest National Laboratory, Richland, Washington, March 2001.
2. A. Strasser, et al., "Control Rod Materials and Burnable Poisons, An Evaluation of the State of the Art and Needs for Technology Development, July 1980," NP-1974, Edison Power Research Institute, November 1981.

3.8 MATERIALS EVALUATION

3.8.1 Material Specification

The TPCTR description of the liner was a "Zircaloy-4" material. Because the function of the liner can be met by most zirconium alloys, the production TPBAR specification for the liner material has been revised to "a zirconium alloy". Commercial ASTM standards are used for procuring and fabricating the 316 SS cladding and end plugs, the zirconium alloy liner and getter, nickel plating of getters, the plenum spring and spring clip. The applicable standards are summarized in Table 3.8-1.

Conclusions

The change in material specification for the liner from Zircaloy-4 to zirconium alloy provides greater flexibility to the TPBAR fabricator in obtaining liners and has no impact on the function of the liner or its compatibility with other internal materials.

3.8.3.1 Material Compatibilities for Normal and Accident Conditions

Cladding Defects

TPBARs are designed and fabricated to the same high quality standards as fuel rods. Therefore, catastrophic failures of TPBARs during Conditions I, II, III, and IV are not expected to occur except for LBLOCA and fuel handling accidents. Any failures under normal conditions are anticipated to be minor fabrication or weld defects, such as pin-hole leaks, with very little likelihood of lithium leaching from the failed rod into the RCS.

Should a TPBAR rupture during reactor operation, it is conservatively assumed that all lithium is immediately leached from the TPBAR. Even with this assumption, power peaks in adjacent fuel due to such cladding defects will not result in a departure from nucleate boiling (DNB) or fuel failure within normal limits of operation. TVA has requested that DOE perform additional tests to provide a more precise understanding of the leach rate and total amount of material that may be leached under these conditions. It is expected that the results of this testing will allow some of the conservatism to be removed from the current assumptions. See Section 3.7.3 for further discussion of failure analyses and the impacts of TPBAR failure.

The lithium from pellet leaching added to the normal lithium content of the RCS has an insignificant effect on the pH. If 100% of the ^{6}Li were leached simultaneously from two adjacent breached TPBARs over three days, core safety limits would not be exceeded, assuming normal operation.

Both the 302 SS plenum spring and the zirconium alloy spring clip are non-reactive with the other TPBAR components. These components are essentially insoluble in reactor coolant and a negligible amount will dissolve into the coolant in the event of a cladding breach.

3.8.3.2 Material Compatibilities following a Large Break Loss of Coolant Accident

The TPCTR noted that limited lithium leaching would occur from a TPBAR in the event of cladding failure. This conclusion was based on limited published information. PNNL recently performed tests for leaching of irradiated absorber pellets under controlled conditions of water composition and temperature similar to what would be expected in a post-Large Break Loss of Coolant Accident (LBLOCA) environment. The pellets did not dissolve, but lithium leaching from TPBAR-like configurations was observed to occur at a rate of <3%/day. Leaching from pellets approached a maximum level of ~50% of the lithium present at the start of leaching.

During a LBLOCA, those TPBARs which experience conditions of high internal pressure coupled with high cladding temperature will rupture. For accident analyses, it is conservatively assumed that up to 50% of the lithium present at the time of the LBLOCA will eventually be leached from ruptured TPBARs. Based on rupture tests performed by PNNL, it is conservatively assumed that no more than one pencil worth (~12") of lithium aluminate absorber pellets may be ejected from the TPBAR at the time of rupture. Analyses demonstrate that the reactor can be maintained in a safe shutdown condition under these circumstances. TVA has requested that DOE perform additional prototypic testing to confirm the conservative assumption of pellet ejection. See Sections 2.15.5.4 and 3.7.3.

Conclusions

The effects of cladding defects have been evaluated and found to be of minimal consequence under conditions of normal plant operation and accident conditions. Analyses have shown that during a LBLOCA, the reactor can be maintained in a safe shutdown condition.

3.10 POST-IRRADIATION EXAMINATIONS FOR THE LTA TPBARS

The TPCTR identified steps to be taken by the Department of Energy (DOE), Tennessee Valley Authority (TVA), and Pacific Northwest National Laboratory (PNNL) to evaluate performance of the Tritium-Producing Burnable Absorber Rods (TPBARs) after the irradiation of the Lead Test Assemblies (LTAs) in cycle 2 of the Watts Bar Nuclear Power Plant (WBN). Following is a summary of monitoring and evaluation that have been performed.

Summary

Based on monitoring performed during the 18-month irradiation cycle in WBN, the TPBARs performed as expected during irradiation. WBN experienced no difficulties during the cycle attributable to the LTAs. Evaluation of the tritium concentrations in the reactor coolant has concluded that the LTA irradiation met its design goal of releasing less than 6.7 Ci/TPBAR/year. Following irradiation and shipping for post-irradiation examination, the TPBARs were intact and undamaged.

Visual examination of the TPBARs in the WBN spent fuel pool (SFP) showed no visible indications of damage to the rods or unusual amounts of corrosion. The TPBARs were easily removed from their host fuel assemblies and reinserted into shipping arrays, thus indicating no unusual growth, bow, or other physical distortion as a result of irradiation.

Nondestructive examinations (NDE) at Argonne National Laboratory-West confirmed that the cladding of all 32 TPBARs remained intact during irradiation and post-irradiation handling and shipping. Neutron radiography and full-length axial spectral gamma scanning confirmed the physical state of the "pencils" and pellet stacks and the physical integrity of internal components.

Analysis of measured rod gas pressures, void volumes, and gas composition confirmed that the TPBAR internal components functioned as designed; that is, the tritium production was as expected and the tritium was contained in the internal components. This qualitative conclusion will be quantified through the destructive examinations to be performed at PNNL.

In summary, the irradiation was completed without any adverse impacts on reactor operation or on the TPBARs. All LTA expectations were met.

Performance During Irradiation and Storage

During the period of time the TPBARs were resident in the WBN core, TVA performed weekly monitoring of the reactor coolant for tritium concentration. As stated in the TPCTR, tritium loss from the TPBARs cannot be specifically measured due to the presence of tritium from other sources in the reactor core. However, an evaluation of the measured tritium concentrations in the reactor coolant concluded that the LTA TPBARs met their design goal of releasing less than 6.7 Ci/TPBAR/year.

In preparation for shutdown of WBN from cycle 2, PNNL requested that TVA take samples of SFP water and measure tritium concentration levels in the SFP prior to and after placing the LTAs in the SFP. This monitoring began two weeks before shutdown, with daily samples taken prior to placing the TPBARs in the SFP and then on a weekly basis for the entire time the TPBARs were in the SFP (March 1999-September 1999). Monitoring indicated no change in tritium concentration during the time the TPBARs were stored.

Nondestructive Examinations

Nondestructive examinations of the irradiated TPBARs are described in section 3.10.2 of the TPCTR. This work was performed by Argonne National Laboratory-West on the Idaho National Engineering and Environmental Laboratory (INEEL) site, beginning in September 1999 and was completed in June 2000. The following nondestructive examinations were performed on all 32 TPBARs at ANL-W.

- Visual examination and photography: All TPBARs were examined visually over the full length in at least two orthogonal orientations. Handling scratches, variations in the oxide appearance, and small amounts of crud deposit were observed. No damage to the cladding was observed.
- Rod length, diameter, and bow measurement: Post-irradiation diameters were approximately the same as pre-irradiation; TPBAR lengths increased approximately 0.1 inch during the irradiation, which was less than allowed for in the design; and maximum TPBAR bow was less than 0.5 inch.
- Axial gamma scanning: Axial profiles of activation products in the TPBARs confirmed the axial power profile for the irradiation. Uniform gamma activities among the TPBARs confirmed the relatively flat distribution of power across the LTAs.
- Neutron radiography: All rods were neutron radiographed over their entire length. These radiographs provided a good "picture" of the axial location and physical state of the pencils and the absorber pellet columns. The radiographs confirmed that the internal components maintained their physical integrity during irradiation and post-irradiation shipping and handling. Cracked absorber pellets were observed but they were maintained in position by the getter and liner. No opening of axial gaps between pencils or between pellets was observed.
- Rod puncture: All TPBARs were punctured; void volume and gas pressure were measured; and gas composition was measured. Analysis of the void volumes, gas pressures, and gas compositions confirmed the predicted tritium production, i.e., tritium production derived from these data agreed with the predicted tritium production. Analysis of the gas composition also confirmed that the internal components performed their function of retaining the tritium.
- An insignificant amount of loose absorber material was found at the bottom of some TPBARs; see Section 3.4.3 for a further discussion.

LTA Destructive Examinations and Results

Four of the 32 LTA TPBARs will be destructively examined by PNNL. Analyses will include assays for tritium, hydrogen, and helium concentrations in individual components, lithium isotopic assay to confirm burnup, and optical metallography and scanning electron microscopy. Confirmation of TPBAR integrity during irradiation was obtained from the NDE results. The destructive examinations will be used to refine design assumptions on TPBAR performance and provide additional benchmark data for design models. The benchmarked design models may be used to support future design modifications and assessments of changing operating conditions on TPBAR performance.

3.11 TPBAR SURVEILLANCE

During TPBAR irradiation, periodic review of the reactor coolant activity measurements taken as part of the plant operation will be performed. Specifically, a review of the tritium activity data for tritium concentration in the reactor coolant system will be measured during normal monitoring of the RCS chemistry as described in the TVA sampling program. See section 2.11.3.

If the reactor coolant tritium concentration should reach a level that indicates a catastrophic TPBAR failure has occurred (see sections 3.5.4 and 3.7.3), a safety evaluation would be initiated to determine any operational restrictions necessary to confirm the results of the plant accident analyses remain valid for the duration of operation under these conditions.

The TPCTR stated that a number of irradiated TPBARs would be shipped to a DOE-specified site for additional post-irradiation examinations after the first production cycle. Based on the performance of the LTA TPBARs, TVA does not foresee a need to perform post-irradiation examinations of additional TPBARs following the first production cycle. From the in-reactor data and non-destructive post-irradiation examinations that have been performed on all 32 LTA TPBARs, there do not appear to have been any unusual performance characteristics. Therefore, unless something unusual is observed in the first production cycle that would question TPBAR performance, this additional testing will not be performed.

Conclusions

A plant surveillance program will be developed by TVA to identify any problems attributable to operation with TPBARs. Unless problems are identified that would require further post-irradiation examinations, TVA does not propose to do additional testing following the first production cycle. There is no impact to personnel or public safety as a result of the elimination of the post-irradiation examinations.

3.12 SUMMARY AND CONCLUSION

The TPBAR as evaluated meets accepted and conservative criteria as a core component in the 17x17 type fuel assemblies inserted in the TVA reactors to be used for tritium production (WBN and SQN-1 and -2). The primary functions of TPBARs located in guide thimble tubes which are not under a CRDM are:

- To absorb neutrons as part of the fuel cycle reactivity control
- To produce and contain tritium

The TPBARs perform their function with acceptable margin to failure during normal operation and in conjunction with design-basis accidents:

- As a core component, the TPBAR does not initiate or increase the severity of an accident but has the potential to affect the radiological consequences of some accidents.
- The TPBARs are compatible with 17x17 assemblies operated in a high power density (up-rated) core of the TVA reactors to be used for tritium production. They are attached to specially designed fuel assembly base plates, are inserted in guide thimbles and are compatible with the fuel assemblies.
- Analysis and comparison with equivalent core component assemblies have shown that the TPBAR will not fail during normal operation and Condition I through IV events, with the exception of a Large Break LOCA and the fuel handling accident. During the Large Break LOCA, TPBARs may fail under conditions of high internal pressure and high cladding temperature.
- The enveloping tritium releases provided as input to the tritium release consequence evaluations are considered conservative.
- TPBARs use materials with known and predictable characteristics in reactor performance and are compatible with the reactor coolant system.
- Detection of excess tritium concentration in the reactor coolant during periodic surveillance will trigger evaluations to ensure safety margins are adequate for continued normal operation or operation during a moderate frequency event.
- The thermal-hydraulic evaluation has shown that TPBARs operate within established thermal-hydraulic criteria.

The evaluation of the production TPBARs incorporates the methodology developed for the TPC TPBARs, including comments raised during the NRC review of the TPCTR, as documented in the TPCTR and the NRC SER.

Table 3.3-1

Production TPBAR Functional Requirements

1. The Production Design TPBAR shall produce up to but not exceed 1.2 grams of tritium per rod while exhibiting acceptable materials performance.
2. The in-reactor tritium release rate for intact Production Design TPBARs shall not exceed a core-wide average of 1000 Ci/1000 rods/yr during normal operation and anticipated operational occurrences.
3. The production Design TPBAR shall not cause adjacent fuel to exceed specified acceptable design limits.
4. The TPBARs shall contribute to reactivity control and power distribution control by use of materials which supplement the negative reactivity of the boron in the coolant.
5. Safe operating temperatures shall be maintained at all times.
6. Tritium release from TPBARs shall not cause radiological regulatory limits to be exceeded. [System requirements that must be met by the TPBAR design in combination with the reactor system.] [*TO BE VERIFIED LATER*]
7. TPBAR failures shall not result in unacceptable core performance.
8. The TPBAR components shall be mechanically compatible with each other and the host fuel assembly.
9. The structural integrity of the TPBAR cladding and end plugs shall be sufficient to perform their functions throughout the irradiation cycle.
10. The mechanical integrity of all internal components shall be sufficient to perform their functions throughout the irradiation cycle.
11. The TPBAR cladding shall remain intact during pool storage and post-irradiation handling prior to arrival at the Tritium Extraction Facility.
12. The TPBAR shall be compatible with the host reactor's fuel assembly design, be a removable component within the assembly, and be located as a stationary element in a guide thimble location.
13. Corrosion-related degradation of TPBAR materials and components shall not occur.
14. The Production Design TPBAR shall be capable of being fabricated in accordance with approved requirements.
15. The unirradiated TPBARs and the unirradiated target assembly must be capable of being transported in accordance with approved requirements.
16. The irradiated TPBARs must be capable of being transported.
17. The TPBAR design shall provide for accountability of each TPBAR.
18. After Irradiation, TPBAR assembly waste must be acceptable for waste disposal.

Table 3.3-2

TPBAR Design Requirements and Assumptions***

Subject Item	TPCRD	SQNTPC Equilibrium Cycle
Maximum tritium production, g/rod	1.2	1.2*
Minimum tritium production, g/rod	0.15	0.15
Core Power Density, W/cm^3	108.04	105.85
GVR limit, rod average**	215	215
Rod internal pressure limit, psia at operating temperatures	3200	3200
TPBAR cladding wall temperature limit, °F @2250 psia system pressure	660	663
Maximum cladding temperature during Conditions I and II, °F	660	663
Bulk boiling temperature in the thimble, °F	652.7	652.7
Maximum cladding structural design temperature, °F	660	663
System pressure, psia	2250	2250
System design pressure, psia	2500	2500
TPBAR life-time, EFPD (nominal without margin)	494	510
Mechanical design life-time, EFPD	520	550
Capacity factor, %	90	100
Tritium release, average, Ci/year	<1.0 per TPBAR	<1000 Ci/1000 TPBARs

* The actual FCD value is 1.175 g/rod with uncertainties applied.
** Gas volume ratio based on theoretical density of lithium aluminate.
*** Use ASME Code stress criteria with Westinghouse generic design stresses for core component rods following the procedure in the Mechanical Design Manual for core rod components.

Table 3.3-3

Significant TPBAR Parameters

Subject Item	TPCRD	SQNTPC Equilibrium Cycle
Maximum Number of TPBARs in core FC/EC	3342/3344[4]	2256
Maximum Number of TPBAR assemblies FC/EC	140	96
Maximum Number of TPBARs per assembly	24	24
TPBAR GEOMETRY & DESIGN		
Cladding OD, in	0.381	0.381
Cladding ID, in. (before coating)	0.336	0.336
Rod OD tolerance, in.	0.0005	0.0005
Rod length, in.	152.37	151.700
Pellet OD, in.	0.303	0.303
Pellet ID, in.	0.223	0.223
^{6}Li loading, g/in. (enriched pellets)	0.030	0.029 & 0.032
^{6}Li enrichment, % (enriched pellets)	25.3	24.46 & 26.99
Enriched pellet stack length (cold), in.	127.5 FC/ 128.5 EC	132
Pellet stack off-set down from centerline, in.	0.50/0.25 FC/EC	0.0 (cold)
Rod back-fill pressure, psia	14.7	14.7
PERFORMANCE PARAMETERS, TPBAR NUCLEAR INPUT		
Guide thimble OD, in.	0.474	0.482
Core Power Density, W/cm^3	108.04	105.85
Average fuel rod power, kW/ft	5.68	5.51
TPBAR average rod power, total, kW (with 8% uncertainty)	5.99	6.86
Peak TPBAR rod power, total, kW (with uncertainties)	8.27	7.80
Average TPBAR rod power, kW/ft with uncertainties	0.498	.572
Total TPBAR power uncertainty factor	1.12	1.145[3]

Notes:
1. Heating rates are for steady state operation.
2. Upper limit tolerance ^{6}Li loading assumed, 4.2% tolerance.
3. Total uncertainty factor is a very conservative bounding value. Consolidation of uncertainties is justified and would reduce the value given above. Future analyses may use a reduced uncertainty, as justified.
4. FC/EC - First Cycle/Equilibrium Cycle.

Table 3.3-3

Significant TPBAR Parameters (Continued)

Subject Item	TPCRD	SQNTPC Equilibrium Core
F_Q with uncertainties	2.5	2.50 x K(z) (including uncertainties
$F_{\Delta H}$ with uncertainties - TPBAR - fuel (max. design)	1.46 1.65	1.52 1.70
Overpower for Condition II, axial average	1.187	1.165
SURROUNDING FUEL ASSEMBLY DESIGN		
Core average axial peak thermal flux, n/cm^2/s,	0.446E14 BOL 0.528E14 EOL	0.3582E14 BOL 0.3578E14 EOL
Axial peak to average neutron flux ratio (F_Z)	1.058 BOL 1.112 EOL	1.177 BOL 1.037 EOL
TPBAR Cladding fast neutron flux, >1 MeV, n/cm^2/s in hot assembly (6,1) location, total flux x 0.24	1.06E14 BOL 1.05E14 EOL	1.05E14 BOL 1.07E14 EOL
TRITIUM PRODUCTION IN FIRST TRANSITION CYCLE (FC) / EQUILIBRIUM CYCLE (EC)		
Tritium production for mechanical and other design assumptions, g	1.2	1.2
Average tritium produced per rod, g	0.856/0.839 FC/EC	0.889
Peak tritium produced per rod (no uncertainty), g	1.089	1.009
Amount of tritium produced per cycle, g	2680/2805 FC/EC	2007 Max TVA Limit
TPBAR average GVR	139/137 FC/EC	138
Axial peak GVR in average rod	156/153.8 FC/EC	147
Axial average GVR in peak rod	174	187
Axial peak GVR in peak rod	195	200
Rod average ^{6}Li burnup, %	45.4/44.2 FC/EC	44.7
Note: Fluxes given for first cycle are larger than equilibrium cycle fluxes		

Table 3.6-2

Evaluation Assumptions

Guide Thimble Tubes Flow Evaluation
1. The fuel assembly coolant temperatures are calculated for a core flow rate reduced by 7.5% bypass flow. This bypass flow rate assumes that the guide thimble tubes contain TPBARs or other core components. Reducing the core flow maximizes the core coolant temperatures and heat transfer into the guide thimble tubes flow.
2. A flow path network of the core was modeled to simulate the appropriate relationship of the guide thimble flow path with that of the adjacent subchannel and the boundary condition of the driving core pressure drop.
3. Fabrication tolerances are used to give the worst case for the analysis being performed.
4. Design tolerances were selected to maximize the guide thimble tube gamma heating.
5. The TPBAR power includes the energy deposited in the water flowing through the guide thimble tubes.
6. The plant is operating at the new rated power level of 3455 at 2250 psia, and nominal T_{in} for boiling considerations.
7. For boiling analysis, a bounding long-term, steady-state axial power shape is used.
8. The TPBAR is operating one pin pitch from the limiting hot rod in the core. The rod adjacent to the thimble tube is modeled as a limiting hot rod reduced in power by the presence of the adjacent TPBAR.
9. The thermal conditions of the flow channels surrounding the guide thimble tubes is obtained from a representative LYNXT code evaluation.
10. Calculations are performed for $F_{\Delta H}$ =1.70 for the limiting hot rod.
Material Temperature Evaluation
11. Overpower conditions, that is, 116.5% power (SQN) is used for maximum TPBAR component temperature calculations.
12. Temperature dependent values of thermal conductivity and thermal expansion coefficient are used
13. One-dimensional, steady-state heat conduction analysis is used in material temperature calculation
14. A bounding total peaking factor, F_Q, is applied for calculation of maximum material temperature. (This bounding factor bounds the plant specific value for both WBN and SQN plants.)

Table 3.8-1

TPBAR Materials and Assembly Specifications

Component	Applicable Material Specification	Associated ASTM Standards
Pressure Boundary		
316 SS Bar Stock	TTQP-1-075, Alloy Grade UNS S31600	ASTM A831/A831 M-95 and ASTM A484/A484 M-94b
316 SS Top and Bottom End Plugs	TTQP-1-079, TTQP-1-080, and TTQP-1-083	ASTM A831/A 831 M-95 and ASTM A484/A484 M-94b
316 SS Seamless Cladding Tubes	TTQP-1-072	ASTM A 771-95
Aluminized Cladding Inner Surface	PNNL-TTQP-1-692	
Absorber Pellets		
Enriched Annular $LiAlO_2$ Pellets	TTQP-1-076	
Getter Tubes and Disks		
Zirconium Alloy Stock Getter Tubes	TTQP-1-073	ASTM B353-95
Zirconium Alloy Getter Disks	TTQP-1-086, TTQP-1-074	ASTM B352-1997
Zirconium Alloy Stock Top and Bottom Spacer Tubes	TTQP-1-073	ASTM B353-95
Nickel Plating	PNNL-TTQP-1-826	ASTM B 689-97
Liners		
Liner Tubes	TTQP-1-077	ASTM B353-95
Springs		
Plenum Springs	TTQP-1-078	ASTM A313-95a
Spring Clips	TTQP-1-089	ASTM B352-97
TPBAR Assembly		
Spacer and Pencil Assembly	PNNL-TTQP-1-688	
Target Rod Final Assembly	PNNL-TTQP-1-690	

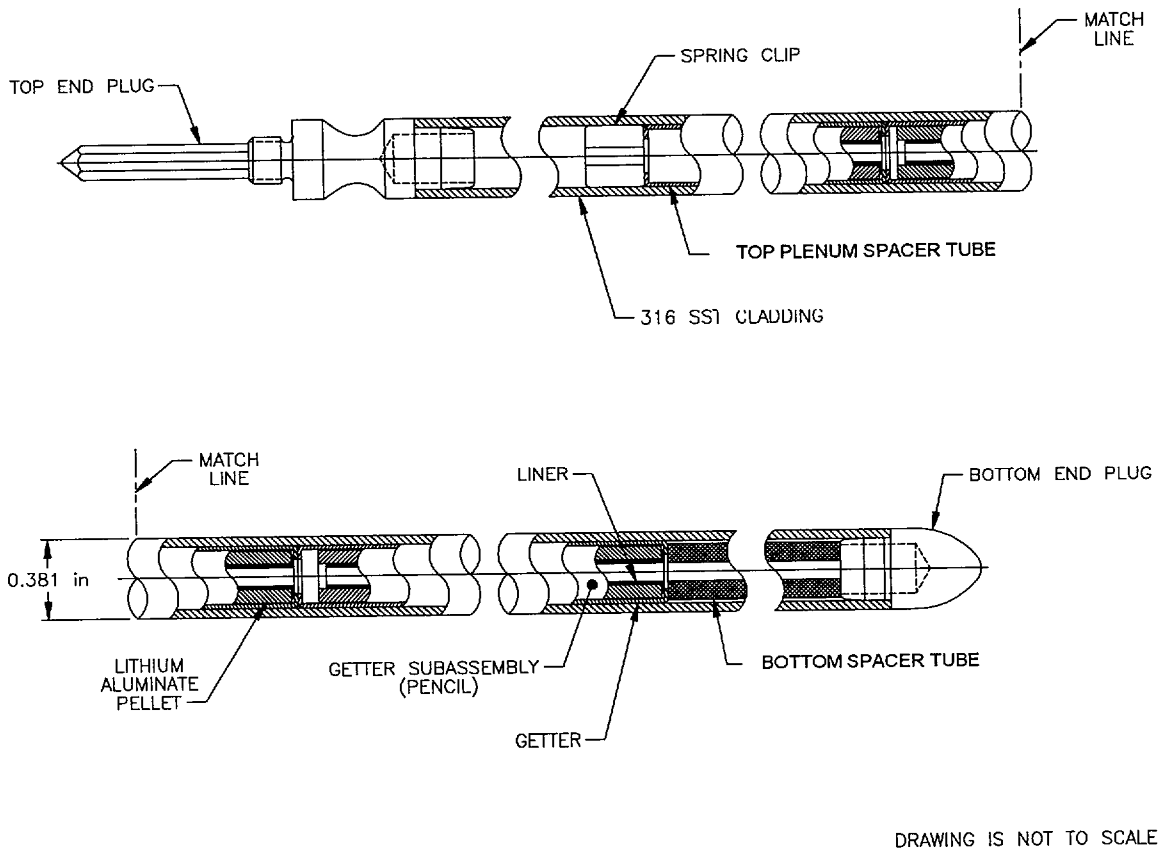


Figure 3.2-1

TPBAR Longitudinal Cross Section

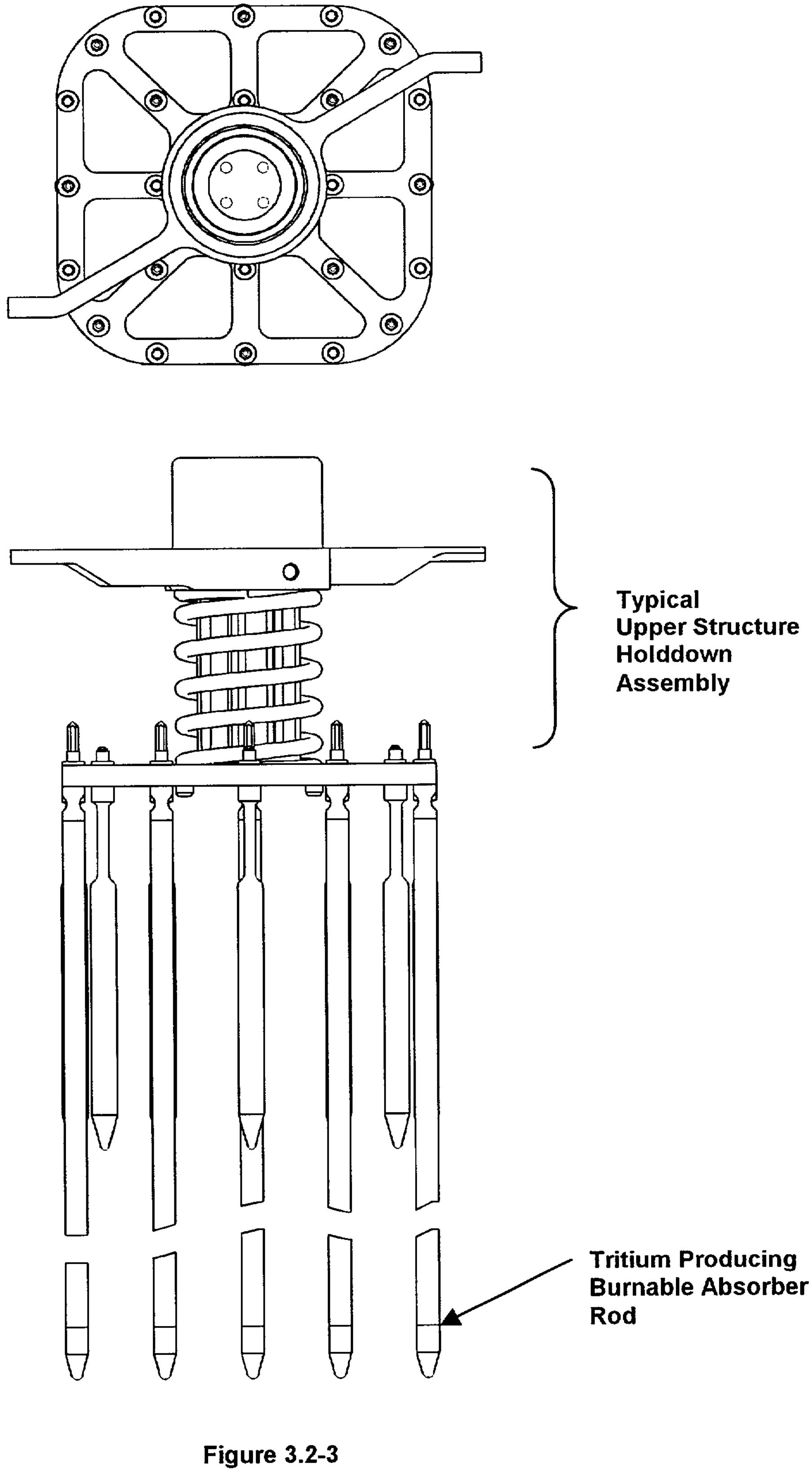


Figure 3.2-3

TPBAR Holddown Assembly

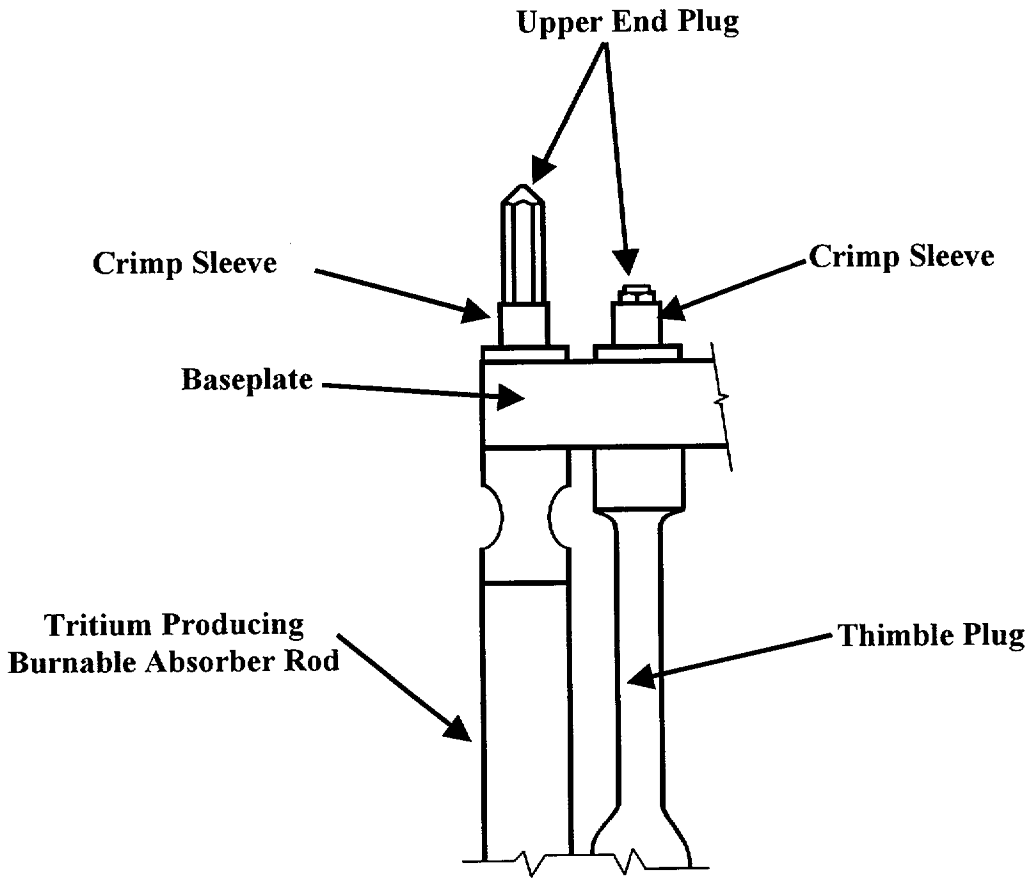


Figure 3.2-4

TPBAR Upper End Plug and Thimble Plug Connections

SECTION 4 PLANT SPECIFIC CONFIRMING CHECKS

The TPCTR identified a number of SRP items for which a plant specific confirming check was recommended. Table 4-1 summarizes the confirming checks performed for SQN Units 1 and 2 which resulted in no impact to the plant. Some of these items are still being confirmed.

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
3.9.1 Special Topics for Mechanical Components	3.9.2	2.3.2	Confirming check recommended for LAR. Response: The pertinent operating parameters (NSSS power, RCS flow, RCS temperatures, steam temperature, feedwater temperature, and steam flow) for the TPC are unchanged from those previously evaluated. Therefore, the existing NSSS design transient curves remain valid.	No Impact
3.9.2 Dynamic Testing & Analysis of Systems, Components & Equipment	3.9.2	2.3.3	Confirming check recommended for LAR. Response: The pertinent operating parameters for the TPC are unchanged from those previously evaluated. The added TPBAR assembly weight, together with the rodlet stiffness, has an insignificant effect on the fuel assembly's dynamic characteristics. The LOCA forces analysis input relative to fuel assembly thimble tube modeling remains bounding for assemblies with or without TPBARs. Therefore, the existing LOCA forces and Flow Induced Vibration evaluations remain applicable.	No Impact

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
3.9.3 ASME Code Class 1, 2 & 3 Components, Component supports, and Core support Structures	3.9.3	2.3.4	Confirming check recommended for LAR, for structural analysis of components. Auxiliary components for spent fuel pit should be reviewed to confirm that design temperatures bound maximum expected temperature. Response: The pertinent operating parameters for the TPC are unchanged from those previously evaluated. The existing NSSS design transient curves remain valid. The existing LOCA forces evaluations remain applicable. Therefore, the TPC has no adverse effect on the component (i.e., steam generator, pressurizer, RCS piping and supports, reactor coolant pumps, reactor vessel, and auxiliary heat exchangers, tanks, pumps and valves) structural analyses.	No Impact
3.9.4 Control Rod Drive Mechanism Design	3.9.4	2.3.5	Confirming check recommended for LAR. Response: The pertinent operating parameters for the TPC are unchanged from those previously evaluated. The existing NSSS design transient curves remain valid. Therefore, the TPC has no adverse effect on the CRDM.	No Impact

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
3.9.5 Reactor Internals Design	3.9.5	2.3.6	Plant specific evaluation recommended for LAR. Response: The T/H evaluation of the Sequoyah reactor internals demonstrated that the core bypass flow, upper head fluid temperature, hydraulic lift forces, and momentum flux are unaffected by the presence of the TPC. The pertinent operating parameters for the TPC are unchanged from those previously evaluated. The existing NSSS design transient curves remain valid. The existing LOCA forces and Flow Induced Vibration evaluations remain applicable. The gamma heating rates that were used in the current evaluations of the baffle-barrel region, the upper core plate and the thermal shield remain applicable. The gamma heating rates seen by the lower core plate increase for the TPC, but an evaluation showed acceptable margins of safety and fatigue utilization factors for all ligaments under all loading conditions. Therefore, the reactor internals will continue to perform their intended design functions for the TPC.	No Impact

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
3.11 Equipment Qualification	3.11.7.2.1 15.5	2.3.7	Confirming check recommended for LAR. Response: *Effects of radiation exposure inside containment will be provided later* Assessments of the mass and energy releases associated with a TPC, for postulated LOCA and secondary system pipe ruptures, demonstrate that they are bounded by the values for a non-tritium producing core.	*TO BE PROVIDED LATER*
4.6 RCCA Drop Time Evaluation	4.2.3	2.4.5	Confirming check recommended for LAR to verify acceptable results. Response: An analysis performed for the TPC design conditions concluded that the TPC has no effect on the RCCA drop time relative to the up-rated SQN core design.	No Impact

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
5.2.2 Overpressure Protection	5.2.2	2.5.2	Plant-specific evaluation of App. G limit (and potential impact on COMS) recommended for LAR. Response: The pertinent operating parameters for the TPC are unchanged from those previously evaluated. In addition, as discussed in Section 1.5.4, the existing reactor vessel integrity analyses, including the reactor vessel Appendix G limits, remain valid for the TPC. Therefore, the existing COMS analyses and setpoints remain applicable for the Tritium Program.	No Impact
5.4.7 Residual Heat Removal System	5.5.7	2.5.4	Plant specific evaluation of the net effect of TPC on RHR System cooling capability is recommended. Response: An analysis has quantified the actual TPC impact on core heat loads at approximately 0.3 MWt. This value represents approximately 1% of the heat load imposed on RHRS during the cooldown period. A review of the RHRS design basis heat load analysis, performed to assess the actual impact of a 1% increase in core decay heat, showed that there is no significant impact on RHRS.	No Impact
6.1.2 Protective Coating Systems	3.8.2 6.2.1	2.6.1	No plant-specific evaluation for LAR if no impact on post-accident EQ conditions for candidate plant. Response: *TO BE PROVIDED LATER*	*TO BE PROVIDED LATER*

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
6.2.1 Containment Functional Design 6.2.2 Containment Heat Removal Systems	6.2.1 6.2.2	2.6.1 2.6.2 2.6.3 2.6.4	Plant-specific confirmation that core stored energy (and, therefore, M/E releases) do not increase is recommended for LAR. Response: A confirming check has been performed which showed that the key safety analysis parameters (moderator density coefficients and shutdown margin) use in the SQN safety analyses for steamline and feedline break M&E releases bound the TPC design values. In addition, the NSSS performance parameters remain bounded. Therefore, the licensing-basis analyses of record for the high-energy secondary-side line breaks remain valid, and the conclusions with respect to M&E releases and the associated pressure and/or temperature response analysis also remain valid for the TPC. A confirming check of the impact of the TPC on the LOCA M&E releases concluded that the vessel temperatures, core stored energy, core pressure drop, and decay heat model used in the LOCA M&E analyses remain applicable for the TPC. Therefore, the current licensing basis analyses remain applicable. There is no adverse impact due to the TPC on the M&E releases to containment.	*TO BE PROVIDED LATER*
6.3 Emergency Core Cooling System	6.3.2.4 6.3.3.15	2.6.1	Confirm no impact on post accident EQ conditions for candidate plant. Response: *TO BE PROVIDED LATER*	TO BE PROVIDED LATER

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
6.5.3 Fission Product Control Systems and Structures		2.6.1 2.15.6	A plant-specific evaluation is recommended for the LAR. Response: The assumed containment design leakage rates, isolation methods and times will remain the same as specified in each of the plant's design basis and will not impact the calculated doses for a design basis LOCA.	No Impact
7.2 Reactor Trip System 7.3 Engineered Safety Features System	7.2 7.3	2.7.2	For LAR, a plant-specific core design will be prepared. If one of the goals is to optimize on fuel usage, safety analysis input parameters could change, requiring a change to the protection system setpoints. Therefore, a review of this area is recommended. Response: Thermal hydraulic studies performed by FRA-ANP conclude that the implememtation of TPBARs in the fuel assembly guide tubes at Sequoayh would have an insignificant effect on RCS flow. It follows that TPBARs would have no effect on RCS temperature or pressure. There is, therefore, no need for a change in reactor trip or ESFAS setpoints and no impact to the core safety limits.	No Impact
7.4 Safe Shutdown Systems 7.5 Information Systems Important to Safety	7.4 7.5	2.7.3	For the LAR, if the candidate plant employs bottom mounted thermocouples, it is recommended that the process measurement effects for post accident monitoring be revalidated with TPBARs accounted for. If the candidate plant does not employ bottom mounted thermocouples, then no plant-specific evaluation is recommended. Response: SQN has top mounted thermocouples, thus no additional evaluation is required for a TPC.	No Impact

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
7.7 Operational Transients/Margin to Trip	7.7	2.7.4	For LAR, a plant-specific evaluation is recommended if: the NSSS performance parameters change , the protection system setpoints change, or the fuel reactivity changes are significant with the TPC. Response: The SQN TPC does not result in changes to the NSSS performance parameters or the protection system setpoints. A comparison of core design reactivities for a typical SQN core design to those for the SQN TPC resulted in the conclusion that there are no significant differences. Therefore, the TPC will not materially affect the plant response for normally expected plant operability transients.	No Impact
Ch. 8 Electric Power	3.11 8.3.1.2.3 8.3.2.2	2.8	Confirm no impact on post-accident EQ conditions for the candidate plant. Response: *TO BE PROVIDED LATER*	*TO BE PROVIDED LATER*
Ch. 10: Steam and Power Conversion System	10	2.10	No plant-specific evaluation is recommended for the LAR, unless the NSSS performance parameters are modified to accommodate the TPC. Response: The NSSS performance parameters are unchanged from those previously evaluated, therefore there are no impacts on the steam and power conversion systems.	No Impact

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
15.1.1-15.1.4 Decrease in Feedwater Temperature, Increase in Steam Flow, and Inadvertent Opening of a steam Generator Relief or Safety Valve.	15.2.10	2.15.1, 2.15.2.5	Confirming check recommended for LAR. If any key input parameters change (as was the case for the reference plant), reanalysis of affected events is recommended. Response: Analytical inputs were examined for these events, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. The FRA-ANP TPBAR reference core designs do not result in a violation of the Doppler analytical limits. The acceptance criteria for these events, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.
15.1.5 Steam System Piping Failures Inside and Outside of Containment.	15.2.13 15.3.2 15.4.2.1	2.15.2.5	Confirming check recommended for LAR. Response: • Section 2, important notes - primary and secondary mass and energy release. Analytical inputs were examined for the steam line break events, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. The acceptance criteria for these events, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
15.2.1-15.2.5 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve, and Steam Pressure Regulator Failure (Closed).	15.2.7	2.15.2.6	Confirming check recommended for LAR. Response: Analytical inputs were examined for the heatup events, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. FRA-ANP Mark-BW fuel design does not exhibit any changes in initial fuel temperature as a result of the TPBAR core design. The acceptance criteria for these events, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impacts.
15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries.	15.2.9	2.15.2.6		
15.2.7 Loss of Normal Feedwater Flow.	15.2.8	2.15.2.6		
15.2.8 Feedwater System Pipe Breaks Inside and Outside of Containment.	15.4.2	2.15.2.6		
15.3.1-15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions.	15.2.5 15.3.4	2.15.2.7		
15.3.3-15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break.	15.4.4	2.15.2.7.3, 2.15.2.7.4, 2.15.6.4		
15.4.2, 15.4.3 Uncontrolled Control Rod Assembly Withdrawal at Power and Control Rod Misoperation.	15.2.2	2.15.2.8		
15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant.	15.2.4	2.15.2.8		

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition.	15.2.1	2.15.2.8.1	Confirming check recommended for LAR. Response: Analytical inputs were examined for this event, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. TPBAR reference core designs do not result in a violation of the Doppler analytical limits. The acceptance criteria for this event, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.
15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature.	15.2.6	2.15.2.8.2	Confirming check recommended for LAR. Response: The SQN Technical Specification requires that all reactor coolant loops be in operation during plant startup and power operation. The event is, therefore, not credible and does not require an explicit evaluation.	No impact.

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position.	15.3.3	2.15.3	Core-specific evaluation recommended for LAR. Response: The possible effects of the implementation of TPBARs at Sequoyah have been evaluated for this accident. The inputs utilized in the analysis of a fuel assembly misloading event remain bounding and conservative. With strict administrative guidelines in place, the probability of a misplacement of the TPBAR clusters or an incorrect ^{6}Li target loading is very low. It has been determined that, even in the unlikely event that the TPBAR clusters or targets are misplaced, the interchange of fuel assemblies or an error in fuel assembly enrichment will result in a bounding local core power or peaking perturbation, making reanalysis of this event unnecessary. In any case, the misplacement of a TPBAR cluster or target misplacement will result in peaking perturbations that are either noticed in the process of startup testing or are of insufficient magnitude to violate design peaking limits in power operation. It is, therefore, concluded that the margin of safety identified in the current licensing analyses reported for the Inadvertent Loading of a Fuel Assembly into an Improper Position event in the Sequoyah FSAR remains unchanged.	No impact.

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
15.4.8 Spectrum of Rod Ejection Accidents.	15.4.6	2.15.2.8.3	Confirming check recommended for LAR. Response: Analytical inputs were examined for this event, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. FRA-ANP Mark-BW fuel design does not exhibit any changes in initial fuel temperature as a result of the TPBAR core design. The acceptance criteria for these events, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.
15.X.X (not in the SRP) Steamline Break with Coincident RCCA Withdrawal at Power.	15.3.7	2.15.2.8.4	Confirming check recommended for LAR. Response: Analytical inputs were examined for this event, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. FRA-ANP Mark-BW fuel design does not exhibit any changes in initial fuel temperature as a result of the TPBAR core design. The acceptance criteria for these events, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.

Table 4-1

TPBAR Impact on Sequoyah (SQN)/LAR Evaluation Results (Continued)

SRP Chapters & Sections	Affected SQN FSAR Sections	DOE Topical Report Section	Evaluation Results	Impact Summary for SQN
15.5.1, 15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory.	15.2.14	2.15.2.9	Confirming check recommended for LAR. Response: Analytical inputs were examined for this event, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. The acceptance criteria for this event, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.
15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve.	15.2.12	2.15.2.10	Confirming check recommended for LAR. Response: Analytical inputs were examined for this event, related to the implementation of TPBARs at Sequoyah. It was concluded that, considering any potential plant design or operational changes associated with the TPBARs, the inputs remain unchanged. The acceptance criteria for this event, therefore, continue to be met and the FSAR conclusions continue to be valid.	No impact.
15.7.5 Spent Fuel Cask Drop Accidents	9.1.4 15.5.6	2.15.1	Confirming check recommended for LAR. Response: The cask handling accidents associated with the production of Tritium involve a Legal Weight Truck (LWT) Cask. Cask handling over the spent fuel pool is prevented by interlocks. In addition, because the crane is considered equivalent single-failure-proof, cask-drop is not considered to be a credible accident.	No Impact

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

I. DESCRIPTION OF THE TRITIUM PRODUCING BURNABLE ABSORBER ROD (TPBAR) CONSOLIDATION

TVA has designed a TPBAR Consolidation Fixture (TCF) to be installed in the cask loading pit for TPBAR consolidation activities. The TCF is quality related in accordance with TVA's NRC accepted Quality Assurance Program. It will normally be stored in the cask lay-down area when not in use. The TCF includes a video monitoring system, lighting, and tools designed to remove TPBARs from their baseplates. The TPBARs are deposited into a consolidation canister (up to 300 TPBARs per canister). The loaded canister is transferred back into the spent fuel pool for short term storage until ultimately being placed into shipping casks for transport off site. The TPBAR consolidation canister loading concept has been successfully demonstrated at Department of Energy's Savannah River Site facility. The completed TCF and tools will be tested prior to delivery and also after installation to verify proper operation prior to actual use.

Consolidation Sequence:

Each tritium core is loaded with certain fuel assemblies containing up to 24 TPBARs attached to a baseplate (TPBAR assembly). The TPBARs then undergo an irradiation cycle. After the core is unloaded to the spent fuel pool during refueling, the irradiated TPBAR assemblies are removed from the fuel and transferred to available storage locations within the spent fuel pool using a burnable poison rod assembly (BPRA) handling tool. Material accountability for TPBAR assemblies is administratively controlled. TPBARs are normally shipped with the new fuel assemblies to the reactor site. TPBAR assemblies that are inserted into once burned fuel are transferred from their storage location into the required fuel assemblies using a BPRA handling tool.

Approximately 30 days after refueling is complete, TPBAR consolidation begins. The canisters (see attached figures) to receive the irradiated TPBARs are transferred into the spent fuel pool, and placed into the consolidation fixture when required. A TPBAR assembly is then withdrawn from its storage location in the spent fuel pool and moved to the consolidation fixture using the TPBAR assembly handling tool suspended from the spent fuel pit (SFP) bridge crane. A TPBAR release tool is then utilized by personnel on the platform to detach individual TPBARs from the baseplate. The TPBAR slides along frame guides, through a funnel and into a roller brake, to limit its velocity, and then into the consolidation canister. The funnel, roller brake assembly, and canister are angled at approximately 15 degrees to enable the TPBARs to stack efficiently into the canister to maximize the loading. Activities take place underwater at a safe shielding water depth.

After TPBARs have been removed from a baseplate, the baseplate and any attached thimble plugs will be removed from the fixture (utilizing a hand held baseplate tool or a TPBAR assembly handling tool suspended from the SFP bridge crane), and placed in storage. The process is repeated until the canister is filled with up to 300 TPBARs. Disposal or storage of the baseplates and thimble plugs will be in accordance with accepted radwaste programs.

The loaded TPBAR consolidation canister is removed and transported to a designated storage position in the spent fuel pool storage rack using the canister handling tool suspended from the SFP bridge crane. The next empty consolidation canister is placed into the consolidation fixture and the process is repeated until all TPBARs irradiated during the fuel cycle have been consolidated. The consolidation fixture is then removed from the cask load pit and stored in the cask lay-down area. Subsequently, a shipping cask is placed into the cask loading pit. The cask is handled by the Auxiliary Building crane in accordance with NUREG-0612 program requirements. The canisters are transferred into the submerged cask. The cask is removed from the cask loading pit, drained of water and decontaminated, packaged and certified for shipment. This shipping process is repeated until all TPBARs irradiated during the past operating cycle have been shipped.

II. <u>SAFETY ANALYSIS</u>

Other than the removal of the TPBAR assembly from a spent fuel assembly, and transport of a loaded canister to and from the designated SFP storage cells, TPBAR consolidation is performed in the cask loading pit area of the SFP. The following topics are evaluated to provide assurance that consolidation activities do not pose a significant hazard to the plant or personnel:

1. Seismic Qualification of the SFP Racks With Loaded Consolidation Canisters

   The spent fuel pool racks have been seismically qualified containing consolidation canisters loaded with up to 300 TPBARs and have been found acceptable.

2. Heat Produced by the Irradiated TPBARs in a Consolidation Canister

   The additional heat produced by TPBARs (approximately 3 watts per rod at 30 days after shutdown) contained in a fully loaded consolidation canister is approximately 900 watts. Slots have been designed in the consolidation canister bottom and sides to provide flow paths for natural circulation cooling of the TPBARs, which will be adequate to help dissipate this small amount of heat.

3. Maintaining Criticality Limits for the Spent Fuel Racks Containing Loaded Canisters

   Analyses were performed to determine the limiting amount of water that can be displaced in order to checkerboard nonfissile bearing components with fresh fuel. These analyses conservatively determined that 75% of water can be safely displaced in empty cells by nonfissile bearing components. Because a fully loaded TPBAR storage canister containing 300 TPBARs displaces approximately 51% of the water in a storage cell, and the displacing material is a strong neutron poison, no additional restrictions are necessary on the location of the TPBAR canister in the spent fuel pool.

4. Fuel Handling and Storage for Assemblies Containing TPBARs

   The weight of a fuel assembly with 24 TPBARs and its hold-down assembly (63 additional lbs for TPBARs) is less than an assembly with a rod control cluster, and therefore is bounded by the current assumed weight of assembly for purposes of analyzing fuel handling and storage facilities. The TPBAR equipped fuel assembly has the same external configuration to interface with the fuel handling/storage equipment. Additionally, this weight is conservative for purposes of defining a NUREG-0612, "Heavy Load."

5. TPBAR Assembly Handling for Consolidation

The weight of a TPBAR assembly is comparable to a burnable poison rod assembly (BPRA). The configuration of the baseplate and TPBAR attachment details are compatible with existing fuel assemblies and the BPRA handling tool. Therefore, the TPBAR assembly can be handled with the existing BPRA tool or any other tooling designed for the BPRA's. A postulated drop of the light weight, base plate with TPBARs, within the spent fuel pool/cask load pit area, is bounded by the analysis of a fuel handling accident damaging an irradiated fuel assembly and 24 included TPBARs.

6. TPBAR Consolidation Canister Handling

Additional precautions are taken in addition to existing plant processes for handling heavy loads to ensure handling of the loaded canister will limit, to an acceptable level, the possibility of damage to no more than 24 TPBARs during handling.

A. In accordance with NUREG-0612, -0554, and ANSI N14.6, the SFP bridge crane and canister lifting device will contain sufficient aspects of the single failure proof criteria to preclude a drop of the loaded canister as delineated below:

   1. The SFP bridge crane is considered equivalent single failure proof with respect to structural integrity in accordance with NUREG-0612 (NUREG-0554) due to the following:

      a. Since the SFP bridge crane has a capacity of 2000 lbs and the weight of the submerged loaded canister is approximately 700 lbs, the crane has safety factors twice the normally required values.

      b. The crane is equipped with redundant high hook limit switches of different designs to preclude two blocking and subsequent structural failure.

   2. The lifting tool is provided with a safety lanyard attached to a hoist trolley to limit canister descent in the fuel pool to such an extent that spilling of the TPBARs out of the open topped canister is prevented. The lanyard is sized to stop the canister from a maximum hook speed of 40 feet per minute. Administrative requirements require that the

safety lanyard be attached to the lifting tool during hoisting when the canister is not engaged in a SFP rack cell, the consolidation fixture holster, or cask by at least 12 inches.

Additionally, analysis has been performed to demonstrate that damage to more than 24 TPBARs contained in a canister is precluded for all credible impact scenarios during canister handling. This analysis does not analyze a fuel assembly falling onto a loaded consolidation canister located in a spent fuel rack. Accordingly, administrative and/or design features will be in place to preclude the possibility of damage to TPBARs loaded into canisters resulting from a fuel handling accident.

3. In accordance with ANSI N14.6 sections for critical loads, the lifting tool is designed to twice the normal safety factors, tested to twice the normally required loads, and inspected utilizing required nondestructive testing methods, thereby rendering it equivalent single failure proof. It will also have a fail-closed safety latch to prevent the tool hook from disengaging from the canister lifting bail.

B. The loaded canister weight and its handling tool is less than that of a fuel assembly and its handling tool. Additionally, due to the design features listed above, the canister descent is limited to an uncontrolled lowering (e.g., a control failure) of a canister at a maximum hoist speed of 40 feet per minute, thereby limiting the kinetic energy to less than that of the fuel assembly during a postulated free-fall fuel handling accident. Therefore, fuel assembly drop accidents in the pool remain bounding with respect to damage to a stored fuel assembly.

7. Potential Damage to the Cask Loading Pit Liner and TPBARs from the Consolidation Fixture Installation and Handling

The consolidation fixture is designed to remain in place in both its use and storage positions during all credible postulated accidents and natural phenomena, precluding damage to other safety-related systems, structures, and components. This seismic category 1(L) design precludes damage to the spent fuel pool liner in the cask loading pit and consolidated TPBARs while in the fixture.

Due to close proximity to spent fuel in the pool, precautions are taken, in addition to existing plant processes for handling heavy loads, to ensure handling of the consolidation platform will limit, to an acceptable level, the possibility of a platform handling event. Accordingly, the handling of the consolidation platform is performed with the 125/10-ton Auxiliary Building crane and is considered equivalent single-failure-proof for this lift due to the following considerations:

A. The platform (or platform sections) weigh substantially less than ½ of the hook capacity of 125 or 10 tons (Note: The platform is handled as a single unit, and in two sections during assembly). Along with other design and administrative features, this crane is considered equivalent single-failure-proof consistent with the requirements of NUREG-0612 and NUREG-0554 for this lift.

B. The lifting devices are designed to the requirements of ANSI N14.6 for critical loads with increased safety factors and load test weights, in addition to the design, fabrication, inspection, and testing contained in Sections 1 through 7 of ANSI N14.6, therefore the lifting devices are considered equivalent single-failure-proof.

8. TPBAR Transport Cask Handling

The aspects of cask handling accidents associated with the production of Tritium are the radiological effects of consolidated TPBARs in a legal weight truck (LWT) cask, and potential interactions between the cask and other safety-related systems, structures and components. No significant hazards to the plant or public are created due to the following considerations:

A. Due to close proximity to spent fuel in the pool, precautions are taken, in addition to existing plant processes for handling heavy loads, to ensure handling of the cask will limit, to an acceptable level, the possibility of a cask handling event. Accordingly, the handling of the LWT cask is performed with the 125-ton Auxiliary Building crane and is considered equivalent single-failure-proof for this lift due to the following considerations:

1. The LWT cask weighs less than ½ of the crane capacity of 125 tons. Along with other design and administrative features, this crane is considered equivalent single-failure-proof consistent with the requirements of NUREG-0612 and NUREG-0554 for this lift.

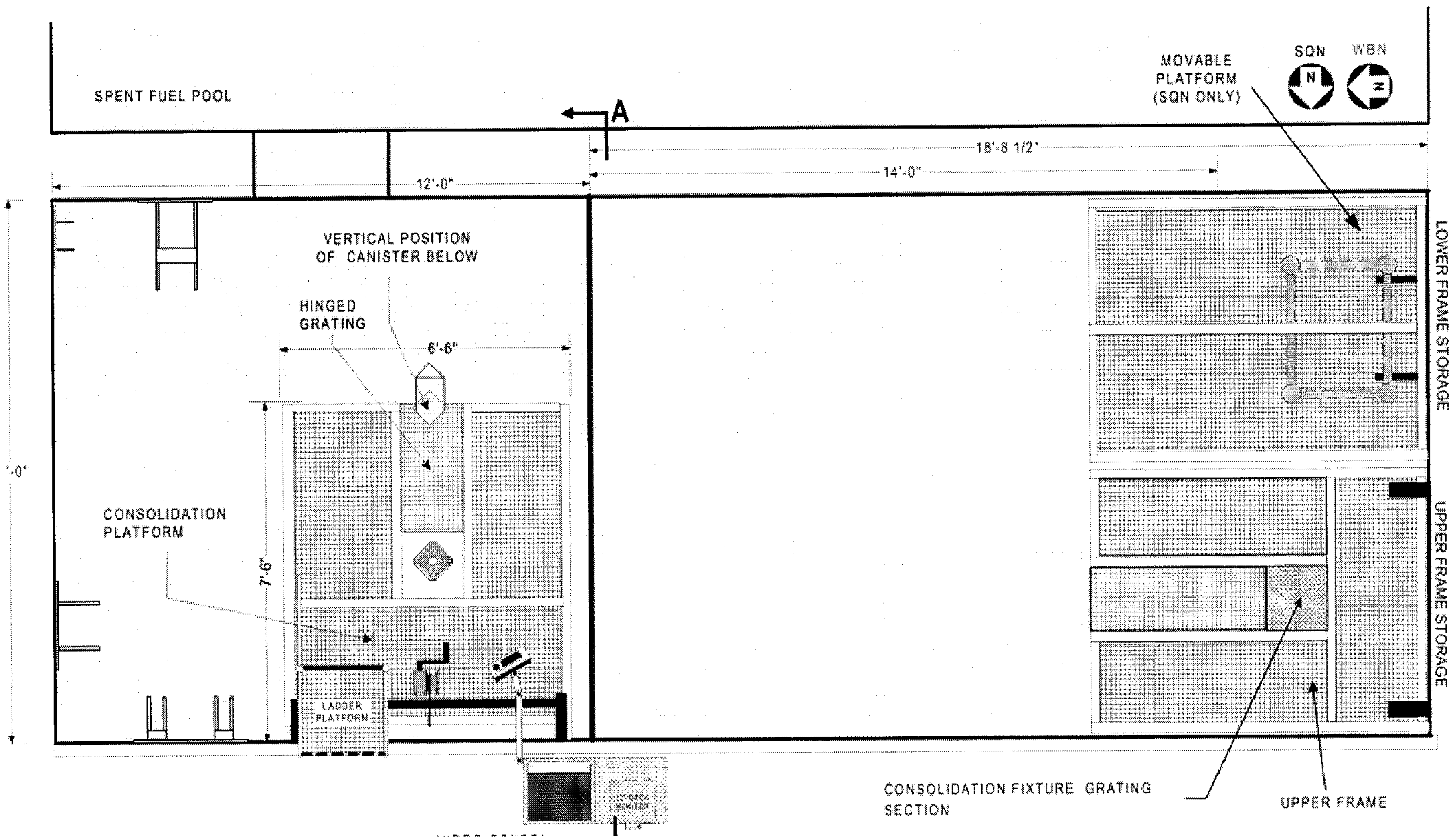
2. The lifting device is designed to the requirements of ANSI N14.6 for critical loads with increased safety factors and load test weights, in addition to the design, fabrication, inspection, and testing contained in Sections 1 through 7 of ANSI N14.6, therefore, the lifting device is considered equivalent single-failure-proof.

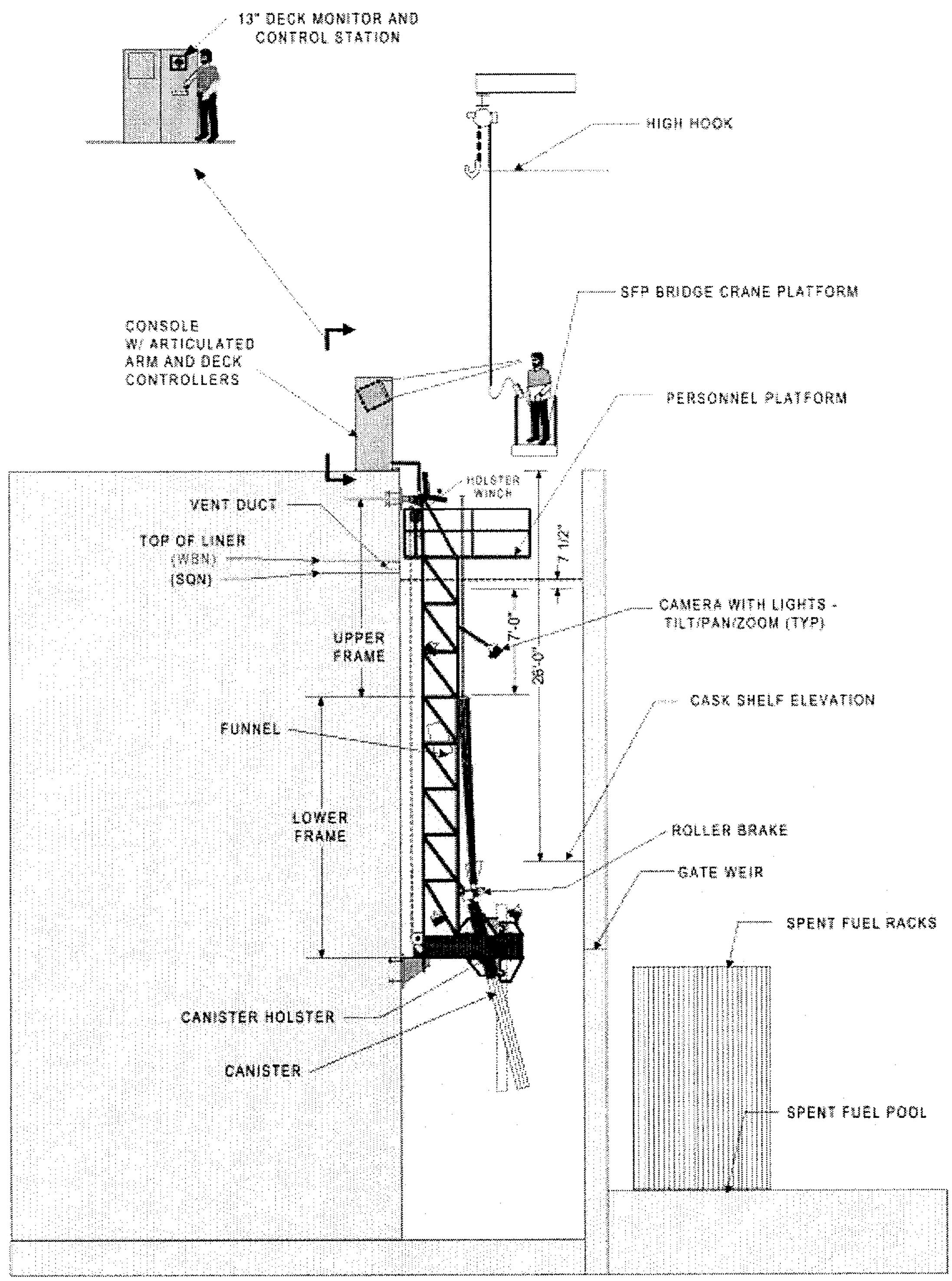
B. All other NUREG-0612 requirements as delineated in response to Generic Letter 81-07 for this crane, such as crane interlocks preventing crane hook travel over the new and spent fuel pools, safe load paths, crane inspection and operator training, etc., remain in force.

9. Worker Radiation Exposure During TPBAR Consolidation Activities

The TPBAR handling and consolidation equipment is designed and configured such that minimum water shielding in the spent fuel pool and cask loading pit is maintained to keep dose rates ALARA (As Low as Reasonably Achievable). Tool design/features prevent inadvertently raising the TPBAR assemblies, loaded canisters or post consolidation baseplates above safe shielding depths.

Personnel will work on a platform 24 inches above SFP normal water level over the deep end of the cask loading pit. The platform is designed to accommodate lead shielding, if required, for personnel protection.


SPENT FUEL POOL
A
MOVABLE PLATFORM (SQN ONLY)
SQN
WBN
N
18'-8 1/2"
14'-0"
12'-0"
VERTICAL POSITION OF CANISTER BELOW
HINGED GRATING
6'-6"
7'-6"
CONSOLIDATION PLATFORM
LADDER PLATFORM
LOWER FRAME STORAGE
UPPER FRAME STORAGE
CONSOLIDATION FIXTURE GRATING SECTION
UPPER FRAME



13" DECK MONITOR AND
CONTROL STATION
HIGH HOOK
SFP BRIDGE CRANE PLATFORM
CONSOLE
W/ ARTICULATED
ARM AND DECK
CONTROLLERS
PERSONNEL PLATFORM
HOLSTER
WINCH
VENT DUCT
TOP OF LINER
(WBN)
(SQN)
7 1/2"
7'-0"
26'-0"
CAMERA WITH LIGHTS -
TILT/PAN/ZOOM (TYP)
UPPER
FRAME
CASK SHELF ELEVATION
FUNNEL
LOWER
FRAME
ROLLER BRAKE
GATE WEIR
SPENT FUEL RACKS
CANISTER HOLSTER
CANISTER
SPENT FUEL POOL

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

I. DESCRIPTION OF SPENT FUEL POOL (SFP) COOLING ANALYSIS METHODOLOGY CHANGES

In order to accommodate earlier off-loading of the core consistent with existing Technical Specification limitations on fuel movement (100 hours), TVA proposes by this submittal to augment its analysis of record, which develops an alternate analysis that increases the maximum allowable SFP decay heat load up to a maximum of 55 MBtu/hr by taking credit for actual (lower) component cooling system (CCS) water temperatures and actual (lower) SFP heat exchanger fouling factors. Although this appears only to be a change in input values, the approach as to how these values are used is different. Therefore, TVA has considered this use as a methodology change.

Because of this change, certain portions of the Updated Final Safety Analysis Report (UFSAR) will require modifications. However, even with these changes, the maximum design basis temperature for SFP remains unchanged and is bounding for higher decay heat loads.

In addition to the above, NRC is aware that SQN Units 1 and 2 have been selected by the Department of Energy (DOE) to provide irradiation services for tritium producing burnable absorber rods (TPBARs) in support of maintaining the nation's tritium inventory. As a result of TPBAR irradiation, there will be a small increase in decay heat loads placed into the SFP. TVA's existing analysis of record utilizes design basis values and bounding fuel discharge scenarios for predicting maximum SFP temperatures. This analysis determines the limiting decay heat loads for the pool. The UFSAR also allows placement of spent fuel into the pool regardless of discharge scenario, provided that the maximum allowable spent fuel heat load is not exceeded. This proposed change will allow TVA to offset the increase in heat load due to planned TPBAR production activities.

II. SAFETY ANALYSIS

TVA proposes to increase the existing SQN SFP heat load limit from its current value of 45.37 MBtu/hr to a range between 45.37 and 55 MBtu/hr. Such a change will compensate for the projected increase in SFP decay heat resulting from off-loading the core during outages as early as 100 hours after shutdown. The change will also compensate for the projected increase in SFP decay heat resulting from planned tritium production activities. The proposed change to the allowable limit will effectively increase the heat load capability of the SFP cooling system up to a new maximum value of 55 MBtu/hr. Exceeding the lower design value is possible by taking credit for actual (lower) fouling of the SFP cooling system heat exchanger, and by taking credit for actual (lower) CCS temperatures. Analyses have been performed that support the proposed change. The results of the analyses show that the maximum spent fuel temperature for single train SFP cooling operation will not be increased, that localized boiling within the hottest fuel assembly will not occur, and that existing design limitations on heat removal systems will not be exceeded.

Current UFSAR Description of SFP Cooling System

UFSAR Section 9.1.3 states that the SFP cooling and cleanup system (SFPCCS) for SQN is sized to handle full core off-loads. The FSAR reflects a limiting value of decay heat that can be placed in the SFP, based on outage specific decay heat analysis performed for each outage. This approach provides a more realistic means (based on a quantitative limit instead of an off-load scenario based limit) of assuring compliance with the maximum allowable design basis decay heat load. Each outage a core specific and real time SFP decay heat assessment is prepared, which considers core operating parameters such as average fuel burn-up, interim trips, and coast-downs, etc., to develop preoutage data for expected core and SFP decay heat. Procedures are in place to assure that at no time during core off-loading activities will the design basis limits of the SFPCCS be exceeded. Compliance with these limiting values provides assurance that, should a train of SFPCCS fail, maximum analyzed temperatures of the SFP and attendant decay heat removal system piping will not be exceeded.

The current UFSAR recognizes that a complete loss of SFP cooling (loss of two trains) would ultimately result in a SFP boiling condition. Mitigation of such an event is provided by multiple sources of makeup water to ensure the fuel storage racks and their contained fuel assemblies are always submerged by at least 10 feet of water. One of the multiple sources of makeup water includes the fire protection system, a safety grade supply of water. SFP heat up rates and time to boil estimates are provided in the UFSAR, Table 9.1.3-1.

Safety Analysis of Proposed Change

The existing SQN SFP heat load limit is 45.37 MBtu/hr. The proposed change to the allowable limit will effectively increase the heat load capability of the SFP cooling system up to a new maximum value of 55 MBtu/hr. This higher allowable heat load is based on an alternate analysis performed utilizing actual system operating parameters. Exceeding the lower design value will only be permitted under consideration of actual fouling of the SFP cooling system heat exchanger, and by taking credit for actual (lower) CCS temperatures.

SFP Heat Exchanger Fouling Factor

The analysis of record utilized a design fouling factor of approximately 0.0005 (hr*ft2*°F/Btu) for both the tube and the shell side fouling. Actual fouling of the SFP heat exchangers has been found to be considerably less than design, with minimal negative trending over a long period of time, based on Sequoyah experience. This experience is consistent with expectations, given that both the CCS and the SFPCCS streams are clean water systems, approaching demineralized water in purity and clarity. Any particulate impurities introduced in the SFP during fuel movement operations would not be sufficient to foul the heat exchangers, as the water velocity in the SFP is very low. Any significant introduction of particulate impurities would adversely affect optical clarity of the SFP requiring fuel movement operations to cease. Therefore, the possibility of sudden fouling of the SFPCCS heat exchangers during fuel off-load operations is not credible.

The conditions required for fouling of the heat exchanger are not present in the SFPCCS. Actual data to date from SQN suggest low fouling rates of the heat exchanger over 20 years without cleaning. The use of this new methodology will require the use of certified Measuring and Test Equipment (M&TE) under written procedures for the determination of heat exchanger fouling factors prior to taking credit for lower fouling factors. Due to the high purity of the coolant and cooled streams, and the proven history to date of low fouling, high fouling rates or other deviations to any established trend are not likely. Analyses performed with less than design fouling indicated significant benefit can be obtained in removing additional heat load from the SFP.

CCS Maximum Water Temperature

The analysis of record utilized design maximum values for CCS temperatures for the cooling medium on the shell side of the SFP heat exchangers. The maximum design temperature for CCS during refueling outages is 95°F. This value, however, is very conservative relative to the actual amount of heat being rejected to the CCS. The design basis for the CCS included significantly higher decay heat loads based on residual heat removal (RHR) system heat loads shortly after shutdown. By the time the core is completely off-loaded, the RHR heat load is essentially zero. By increasing the flow of essential raw cooling water (ERCW) to the CCS heat exchanger to its design maximum allowable flow, CCS maximum temperature can be decreased to values less than the 95°F design value, even when considering design ERCW temperature and design fouling of the CCS heat exchanger.

Results of Alternate Analysis

By performing multiple analyses of SFP thermal performance at varying fouling factors from 0.0005 to 0.0001 (hr*ft2*°F/Btu) and decreased CCS temperatures, a series of curves have been developed to provide operator guidance for an increase in allowable SFP decay heat. Analyses were performed for the limiting case of single train operation, in which the allowable design heat load was increased up to a maximum without exceeding the maximum design SFP temperature. Final curves of allowable decay heat verses CCS temperature and SFP heat exchanger fouling were developed, which included a margin to account for inaccuracy inherent in reading graphs, and to add additional modeling conservatism. The curves provide design guidance for use in procedures, which allow increased decay heat load in the SFP based on actual values for CCS temperature and SFP heat exchanger fouling.

A complete loss of SFPCCS (two trains) could result in a SFP boiling event in less than 3 hours. The alternate analysis has shown that even with higher allowable decay heat loads in the SFP, adequate sources of makeup water exist to allow sufficient time (weeks) to mitigate such an event, without reducing the SFP water level to unacceptable levels (10 feet above fuel storage racks). While the alternate analysis has shown a decrease in the time to react to a complete loss of SFP cooling, the resulting time available to mitigate such an event remains acceptable. Additionally, the analyses for loss of cooling events all considered steady state heat loads from the fuel. There is low probability of reaching an unacceptable level of coolant in the SFP. A loss of two trains must first be postulated coincident with maximum heat load in the SFP, after which there exists a time period of over several weeks to restore cooling. The actual heat load in the SFP would decay during this time period to levels which would not support a boil-off rate in excess of makeup capability. Multiple sources of makeup water (one qualified) exist to maintain and restore SFP level. Therefore, based on these factors, assurance is provided that the proposed change will not unacceptably decrease any margin of safety associated with SFPCCS operation or storage of spent fuel.

As a result of higher allowable heat loads in the SFP, the previous localized boiling analysis of record has been impacted. The localized boiling analysis was re-performed consistent with existing analysis methodologies except the rack and pool area were modeled using a three-dimensional nodalization, instead of two dimensional. The inputs to the analysis were revised to be consistent with the higher proposed maximum allowable decay heat value (55 MBtu/hr). The results of the analysis show that while the margin to localized boiling has decreased, localized boiling within a rack will not occur. The analysis specifically concluded that:

1) the maximum local water temperature in the fuel storage racks was less than the local saturation temperature of the water, and

2) the maximum fuel clad temperature, while greater than the local water saturation temperature, would not result in departure from nucleate boiling (DNB), and that fuel cladding integrity would be maintained.

The following is a tabulation of specific SFP design values and parameters for both the existing design limiting conditions and the proposed values based on the alternate analysis:

SQN SPENT FUEL POOL DESIGN PARAMETERS		
Parameter	**Existing Design Value**	**Proposed Value (Alternate Analysis)**
Maximum Allowable Decay Heat Load	45.37 MBTU/Hr	45.37 - 55 MBTU/Hr See Note 1.
SFPCCS Flow	2300 GPM per Hx	2300 GPM per Hx
CCS Flow	3000 GPM per Hx	3000 GPM per Hx
Allowable Tube Plugging	5%	5%
Tube-Side Fouling (hr*ft^2*°F/Btu)	0.000575	0.0005 - 0.0001
Shell-Side Fouling (hr*ft^2*°F/Btu)	0.0005	0.0005 - 0.0001
Maximum CCS Temperature	95°F	95 - 80°F (Note 1)
Maximum SFP Temperature (1-Train)	183°F	183°F
Maximum SFP Temperature (2-Train)	144°F	144°F
Average Time to SFP Boiling	2.64 Hours	1.14 Hours
Average SFP Heat-Up rate	10.98°F/Hr	25.35°F/Hr
Average Boil-Off Rate	103 GPM	118.2 GPM
Time until only 10 feet of water over racks - without makeup	30 Hours	25.7 Hours
Time until only 10 feet of water over racks - with 103 gpm makeup	See Note 2	See Note 2
Margin to Localized Rack Boiling	4.8°F	3.5°F
DNB at maximum heat load and maximum SFP temperature.	No	No

Notes:

1. The range of values represent allowable heat loads based on specific combinations of heat exchanger fouling between 0.0005 and 0.0001 (hr*ft^2*°F/Btu) and actual CCS temperatures between 95 to 80°F.
2. Analysis has shown that SQN has a qualified source of makeup water of 103 GPM, therefore the 10 feet above rack level is never reached for the boil-off rates determined.

Summary

The SFPCCS has adequate capacity and cooling margin to perform its safety and nonsafety functions with the additional heat loads imposed by the proposed change to allow commencement of core off-loads as early as 100 hours, consistent with specific requirements regarding SFP heat exchanger fouling and CCS temperature. The SFPCCS system can also accommodate the additional SFP heat loads imposed by tritium production activities. The increased heat load in the SFP can be safely removed via associated heat removal systems within existing design basis analyses.