S64 020719 800

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

July 19, 2002

TVA-SQN-TS-00-06 10 CFR 50.90

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 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-06, REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NOS. MB2972 AND MB2973)

References:

1. TVA letter to NRC dated September 21, 2001, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Revision of Instrumentation Measurement Range, Boron Concentration Limits, Reactor Core Limitations, and Spent Fuel Pool Storage Requirements for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. 00-06"

2. NRC letter to TVA dated June 6, 2002, "Sequoyah Nuclear Plant (SQN), Units 1 and 2 - Request for Additional Information on Technical Specification Change No. 00-06, Tritium Production Cores (TAC Nos. MB2972 and MB2973)"

3. TVA letter to NRC dated July 3, 2002, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2

DD30

- Technical Specification (TS) Change No. 00-06, Response to Request for Additional Information (RAI) (TAC Nos. MB2972 and MB2973)"

TVA is submitting a revised response to the Referenced 3 letter to apply the appropriate withholding request for proprietary information. This submittal replaces the Referenced 3 letter in its entirety. TVA requests that the Referenced 3 letter be returned upon receipt of this letter.

TVA submitted TS Change 00-06 to NRC by the Referenced 1 letter to propose changes to the SQN TSs that will accommodate the production of tritium. This letter provides the responses to NRC questions contained in the Referenced 2 letter regarding proposed TS Change 00-06.

Enclosure 1 to this letter provides responses to the NRC RAI in the Referenced 2 letter. There are no new commitments contained in this letter and the proposed TS change in the Referenced 1 letter is not altered by the enclosed responses.

Attachment 1 to Enclosure 1 contains information proprietary to Westinghouse. Accordingly, Enclosure 2 includes Westinghouse Application for Withholding Proprietary Information from Public Disclosure, and accompanying Affidavit CAW-02-1537 signed by Westinghouse, the owner of the information. Also included are a Proprietary Information Notice and a Copyright Notice.

The above affidavit sets forth the basis on which the requested information may be withheld from public disclosure by the Commission, and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.790 of the Commission's regulations. Accordingly, TVA requests that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790.

Correspondence regarding the proprietary aspects of the Westinghouse information listed above, the Copyright Notice, or the supporting affidavit, should reference Westinghouse letter CAW-02-1537 and be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Attachment 3 to Enclosure 1 contains information proprietary to Holtec International. Accordingly, Enclosure 3 includes Holtec International's Application for Withholding Proprietary Information from Public Disclosure, and an accompanying Affidavit signed by Holtec International, the owner of the information.

The above Application and Affidavit set forth the basis on which the requested information may be withheld from public disclosure by the Commission, and addresses with specificity the considerations listed in paragraph (a)(4) of 10 CFR 9.17 and paragraphs (a)(4) and (b)(1) of 10 CFR 2.790 of the Commission's regulations. Accordingly, TVA requests that the information which is proprietary to Holtec International be withheld from public disclosure in accordance with 10 CFR 9.17 and 10 CFR 2.790.

Correspondence regarding the proprietary aspects of the Holtec International information listed above or the supporting affidavit, should be addressed to K. K. Niyogi, Director of Consulting Division of Holtec International, Holtec Center, 555 Lincoln Drive, West Marlton, New Jersey 08053.

This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this response, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the forgoing is true and correct.

Executed on this 19th day of July 2002.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Enclosures

JDS:KCW:PMB
Enclosures
cc (Enclosures):

Mr. Lawrence E. Nanney, Director (w/o Enclosures)
Division of Radiological Health
Third Floor
L&C Annex
401 Church Street
Nashville, Tennessee 37243-1532

Mr. Frank Masseth
Framatome ANP, Inc.
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

JDS:KCW:PMB
Enclosures
cc (Enclosures):

Mr. Lawrence E. Nanney, Director (w/o Enclosures)
Division of Radiological Health
Third Floor
L&C Annex
401 Church Street
Nashville, Tennessee 37243-1532

Mr. Frank Masseth
Framatome ANP, Inc.
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

R. J. Adney, LP 6A-C
J. L. Beasley, OPS 4A-SQN
M. J. Burzynski, BR 4X-C
M. H. Dunn, ET 10A-K
D. L. Koehl, POB 2B-SQN
J. E. Maddox, LP 6A-C
NSRB Support, LP 5M-C
R. T. Purcell, OPS 4A-SQN
J. R. Rupert, LP 6A-C
J. A. Scalice, LP 6A-C
K. W. Singer, LP 6A-C
WBN Site Licensing Files, ADM 1L-WBN
EDMS, WTC A-K (Re: S64 010921 800, L44 020612 001,
S64 020703 800)

i:License/TS Submittal/TSC 00-06 Revised RAI Response

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
TECHNICAL SPECIFICATION (TS) CHANGE 00-06

RAI Question 1:

In Sections 1.5.3, 2.4.3 and 2.4.4 of the Sequoyah Nuclear Plant (SQN) Topical Report, the licensee states that compliance with departure from nucleate boiling (DNB) criterion was demonstrated through evaluations performed using standard U.S. Nuclear Regulatory Commission (NRC)-approved reload analytical methods. Please provide a summary of the results of these evaluations with respect to DNB ratio (DNBR) margins and bypass flow for the tritium producing burnable absorber rod (TPBAR) core. Please provide a comparison of DNBR margin and bypass flow for cores with and without TPBARs. Also, discuss any DNBR penalties associated with the TPBARs.

Response:

The evaluations discussed in Sections 1.5.3, 2.4.3, and 2.4.4 of the SQN Topical Report reflect the application of Framatome-Advanced Nuclear Power's (ANP's) standard reload analysis methods to evaluate SQN tritium production core (TPC) designs. Those evaluations involved the application of DNB based maximum allowable peaking (MAP) limits, generated using LYNXT, in a typical reload-type core power distribution analysis, performed in accordance with the reference.

The effects of the TPBARs on core bypass and local power peaking were addressed in the development of the MAP limits and in the development of peaking augmentation factors used in the core power distribution analysis. The effect of the TPBARs on core bypass was addressed through a hydraulic sensitivity analysis which evaluated TPBAR cores with varying mixes of TPBAR assemblies, thimble tubes, and open guide tubes. The results of this sensitivity analysis showed that the 7.5% total core bypass fraction that has been traditionally applied to the SQN licensing analyses will continue to apply to the TPBAR cores. The sensitivity analysis showed that the various core configurations, both with and without TPBARs, would exhibit guide tube bypass fractions that varied by less than 0.5%, and in no case would the total core bypass exceed the 7.5% value already in place for SQN. Therefore, the MAP limits, which are generated using the bounding total core bypass assumption, are not impacted by the presence of the TPBAR assemblies.

The effects of local power spikes that result from axial gaps between TPBAR pencils were also accounted for in the power distribution analysis. Again, predicted allowable peaks from the LYNXT based MAP limit analysis were used to develop allowable peaking reduction factors that accounted for the effect of the local power spikes on DNB. These MAP limit reduction factors, which ranged in value from approximately 0.3% to 4%, were assessed as part of the core power distribution analysis.

As explained in the response to Question 4, Framatome-ANP attempts to preserve 7% peaking margin to the DNB MAP limits for standard reload core designs. Margin to $F_{\Delta H}$ limits is computed as margin to initial condition DNB MAP limits, as described in the reference.

For TPCs, a similar design guideline was used as a target for the design. The following table shows a comparison of minimum steady-state (SS) and initial condition (IC) DNB peaking margins achieved by four recent standard reload core designs to those achieved by the 96-feed first transition and equilibrium cycles analyzed for the SQN Topical Report. The table shows that the DNB peaking margins preserved for the TPCs are similar to those preserved in recent standard reload core designs for the SQN units.

Minimum Margin to SS-DNB and IC-DNB MAP Limits,
100% Rated Thermal Power (RTP)

Core Design	Margin to SS DNB MAP Limit (%)	Margin to IC DNB MAP Limit (%)
SQN-Unit 1 Cycle 12	7	9
SQN-Unit 2 Cycle 10	5	8
SQN-Unit 2 Cycle 11	7	8
SQN-Unit 2 Cycle 12	7	8
SQN 96-Feed 1st Transition TPC	7	9
SQN 96-Feed Equilibrium TPC	5	8

Reference:

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

RAI Question 2:

In Section 2.4.3 of the SQN Topical Report, the licensee lists the following items as being significant differences between the

SQN design as compared to the generic tritium production core evaluated in the NRC Safety Evaluation Report (SER) (NUREG-1672):

a. *SQN assumes a feed batch of 96 Mark-BW fuel assemblies instead of 193 and 140 VANTAGE+TM fuel assemblies. This represents a batch size larger than current SQN reload cores.*

b. *Two ^6Li concentrations are used instead of one. Concentrations slightly higher (0.032 gm/in) and slightly lower (0.029 gm/in) than that in the generic Tritium Production Core Topical Report (TPCTR) analysis (0.030 gm/in) were used.*

c. *A singular, longer ^6Li 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 lead test assemblies used a 142-inch length.*

d. *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.*

e. *Burnable poison rod assemblies containing B_4C-Al_2O_3 pellets were used on the periphery for fluence control in the equilibrium fuel cycle instead of TPBARs.*

f. *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.*

g. *No fuel rod enrichment zone loading was employed except for fuel rods containing gadolinia.*

For each of the above differences please summarize the technical justification for the difference, and discuss how acceptance criteria of Standard Review Plan (NUREG-0800) Section 4.3 are satisfied considering these differences.

Response:

The acceptance criteria for Standard Review Plan (SRP) (NUREG-0800) Section 4.3 are based upon the requirements of 10CFR50, Appendix A, General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28. They specifically address the acceptability of the core power distribution, the adequacy of the reactivity coefficients, the description of the control requirements, control rod patterns and reactivity worths and the acceptability of the analytical methods and data. As indicated, some of the specific features of the SQN TPBAR design and fuel cycle presented in Section 2.4.3 of Topical Report BAW-10237, Revision 1 are different from those evaluated in NUREG-1672. Section 2.4.3 of Topical Report BAW-10237, Revision 1 addresses how each of the acceptance criteria in NUREG-0800 Section 4.3 were met, or describes the plant changes or TSs that were needed for the requirements to be met. Each of these design parameters was incorporated into the CASMO-3/ Nodal Expansion Method Optimized (NEMO) (References 1 and 2) core simulator model that was used to evaluate each of the SRP Section 4.3 criteria. Specifically Section 2.4.3 of BAW-10237, Revision 1 discusses the acceptability of the methodology, fuel burnup, reactivity coefficients, control of power distribution, maximum control reactivity insertion rate, shutdown margins, and xenon stability. Below is a further discussion on the selection of the design parameters and how they were incorporated into the fuel cycles that were evaluated and summarized in Section 2.4.3 of BAW-10237, Revision 1.

a. Batch Size: Batch sizes are normally a variable for each fuel cycle design. Batch sizes are selected to meet the cycle energy requirements while satisfying other safety and licensing criteria. In the case of TPC an additional cycle requirement is the production of tritium. Batch size will be selected to meet both the cycle energy goals and the strategic tritium production goals while satisfying applicable safety and licensing criteria. While batch size may vary from one cycle to the next, cycle specific reload safety evaluations ensure that all safety analysis criteria are met.

   For the generic tritium production core evaluated in NUREG-1672, the objective of the core designers was to maximize tritium production without consideration for constraints such as availability of spent fuel storage. For the SQN TPC topical, the maximum number of feed assemblies evaluated was limited to 96. A 96-feed assembly core was selected for evaluation because it provided for adequate levels of tritium production, minimized changes to neutron fluence through the reactor vessel, and limited spent fuel generation and storage costs. Actual implementation of TPCs at SQN may use feed batch sizes of equal to or less than 96 fuel assemblies.

b. Lithium Concentrations: The use of multiple lithium concentrations allows the fuel cycle designer greater flexibility in power peaking control. Pacific Northwest National Laboratory (PNNL) has justified the use of lithium concentrations between 0.028 and 0.040 grams/inch in TPBARs (Reference 3). The lithium concentrations are explicitly modeled by the NEMO core simulator; therefore, their effects are considered in each phase of the reload safety evaluation.

c. Lithium Poison Column Length and Axial Position: Normally poison lengths are selected to provide acceptable axial power shapes through core life such that adequate margin can be maintained to safety and operating limits. Current SQN fuel cycles use burnable absorbers that are centered axially with respect to the fuel stack.

d. Gadolinia vs. Integral Fuel Burnable Absorber (IFBA): The integral absorber currently offered by Framatome-ANP and used by SQN in current SQN fuel cycle designs is gadolinia. While gadolinia has different characteristics and implementation than IFBA, e.g., it remains for a longer duration than IFBA and is used in fewer fuel rods, the purpose is the same in the TPC cores. Gadolinia, just like IFBA, is used to supplement the TPBARs lithium absorber to provide sufficient soluble boron control at beginning of cycle and power peaking control. The gadolinia rods are explicitly modeled by the NEMO core simulator; therefore, their effects are considered in each phase of the reload safety evaluation.

e. Burnable Poison Rod Assemblies (BPRAs): BPRAs were used to control the reactor vessel fluence for the 96-feed cycles. For lower feed batch sizes it is anticipated that the use of BPRAs will not be needed for vessel fluence control. Framatome-ANP BPRAs have been successfully used in several SQN cores and in other Westinghouse reactors (e.g., North Anna, Surry) as well. The BPRAs are modeled by the NEMO core simulator; therefore, their effects are considered in each phase of the reload safety evaluation.

f. Numbers of TPBARs per Cluster: The total number of TPBARs in a fuel cycle will be limited by the energy production requirements and the maximum allowable fuel enrichment. The number of TPBARs per cluster is selected so as to allow the maximum tritium production possible while maintaining the best core power distribution control possible. To distribute the power evenly across the core the number of TPBARs per cluster will vary. The TPBARs are modeled by the NEMO core simulator; therefore, their effects are considered in each phase of the reload safety evaluation.

g. Enrichment Zone Loading: With the lower feed batch size used, it was not necessary to axial zone load the enrichment of urania fuel rods for power peaking control in the 96-feed cores. Zone loading of the uranium rods can be counterproductive to maximum tritium production goals and therefore would not be used unless needed for power peaking control. As noted, the enrichment of the gadolinia-urania fuel rods is reduced to accommodate the slightly more restrictive limits on these rods. In addition the ends of the gadolinia rods, beyond the poison column, will have low wt% U-235 pellets. The reduced enrichments in the gadolinia fuel rods are modeled by the NEMO core simulator; therefore, their effects are considered in each phase of the reload safety evaluation.

References:

1. STUDSVIK/NFA-89/3, "CASMO-3 - A Fuel Assembly Burnup Program," Studsvik AB, Nykoping, Sweden, November 1989.

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

3. TTQP-1-116, Revision 8, "Production TPBAR Inputs for Core Designers," Pacific Northwest National Laboratory, Richland, Washington, April 2001.

RAI Question 3:

On page 2-6 of the SQN Topical Report the licensee discusses changes to the CASMO-3 and NEMO computer codes. The licensee modified the cross section libraries and the cross section generation process in the CASMO-3 and NEMO computer codes to include isotopes important for TPBAR cores (tritium, helium and lithium isotopes). Please discuss how these code changes were demonstrated to be accurate through either a verification or benchmarking program and verify that any code changes made are within any code restrictions or limitations identified in the NRC staff SERs for these codes. Also, justify the use of ENDF-B/V rather than ENDF/B-VI (BNL-NCS-17541) cross section libraries.

Response:

The SER in the NEMO Topical (reference) states, "The approved NEMO code may be used only for the range of fuel design configurations and core design parameters that were verified in the topical report. Any fuel or core designs with significant differences that might be introduced must be further validated by B&W or by the licensee."

The overall neutronic model of Li^6 and He^3 in the guide tubes was validated for TPBARs by checking the overall model, the cross section generation, the agreement with Monte Carlo Neutron Photon (MCNP), and comparisons to an operating reactor with TPBARs. Li^6 and He^3 are both 1/v absorbers and the energy dependent neutron reaction cross sections ($n+Li^6\rightarrow$á$+T$ and $n+He^3\rightarrow p+T$) are nearly identical in shape versus energy (not magnitude) to B^{10} ($n+B^{10}\rightarrow$á$+Li^7$). The topical benchmarks include configurations with B^{10} in the guide tubes for the critical experiments, multi-assembly problems, and operating pressurized water reactors (PWRs). The existing topical validates that the current methods can adequately treat a 1/v absorber in the guide tube locations.

The cross section generation was checked. The cross sections for B^{10} were created for CASMO-3 from ENDF/B-V and yielded equivalent cross sections to the existing B^{10} cross sections in CASMO-3. The Li^6 and He^3 cross sections were created for CASMO-3 using the same technique. The ENDF/B-V library was chosen to avoid possible inconsistencies because it was judged to be more representative of the base cross sections in CASMO-3 than ENDF/B-VI. In addition, the cross sections in V and VI for those reactions listed above were reviewed and no significant differences were noted that would affect calculations for PWR applications.

The CASMO-3 model with the new library was validated with comparisons to MCNP assembly calculations containing TPBARs and the standard burnable poison product. The range of reactivity differences between CASMO-3 and MCNP for TPBARs (0.31-0.40 %Δk/k) was less than the range calculated for the standard burnable poison product (0.04-0.79 %Δk/k).

WBN Cycle 2 contained four Lead Test Assemblies (LTAs) each containing eight TPBAR pins and was simulated with NEMO. Predicted power distributions were compared to measured values using the reaction rates. The mean and standard deviation for the assembly average reaction rates for the LTAs were -0.3% and 1.6%, respectively. The mean and standard deviation for the assembly average reaction rates for the measured locations in Cycle 2 were 0.2% and 1.5%, respectively. The range of means and standard deviations for the assembly powers in the NEMO Topical were -0.5 to 0.6% and 1.2 to 2.9%, respectively (reference). Hence, the reaction rate predictions for the TPBAR LTAs were statistically similar to the existing database.

All these results demonstrate the CASMO-3/NEMO will have similar accuracies with TPBARs as with our current core designs.

Reference:

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

RAI Question 4:

On page 2-8 of the SQN Topical Report the licensee states that for core power distribution control, acceptable margins to the F_Q and $F_{\Delta H}$ peaking limits are maintained such that the design bases continue to be met. Please quantify the margins remaining to the peaking limits for a tritium producing core at SQN.

Response:

For standard reload core designs, TVA sets a design guideline for minimum peaking margin to the loss of coolant accident (LOCA) F_Q limit of 7% peaking margin at axial flux difference (AFD) limits of -13% and +7% for operation at RTP. Framatome-ANP applies the same design guideline to DNB peaking margins. Margin to $F_{\Delta H}$ limits is computed as margin to the initial condition DNB MAP limits, as described in the reference.

Depending on the variation from one core design to the next, the 7% target may or may not be met. A variation of 7% ± 2% is typical based on normal changes in core design from one fuel cycle to the next.

For the TPCs analyzed in the SQN Topical Report, Framatome-ANP attempted to preserve 7% margin at the AFD limits. The following table shows a comparison of minimum LOCA F_Q and IC DNB peaking margins achieved by four recent SQN standard reload core designs to those achieved by the 96-feed first transition and equilibrium cycles analyzed for the SQN Topical Report.

Minimum Margin to LOCA FQ Limit at AFD Limits, 100% RTP

Core Design	Margin to LOCA FQ Limit (%)	Margin to IC DNB MAP Limit (%)
SQN-Unit 1 Cycle 12	6	9
SQN-Unit 2 Cycle 10	5.5	8
SQN-Unit 2 Cycle 11	6	8
SQN-Unit 2 Cycle 12	4.6	8
SQN 96-Feed 1st Transition TPC	7	9
SQN 96-Feed Equilibrium TPC	7	8

From this comparison, it is observed that margins to the peaking limits for the two SQN TPCs were similar to those for standard reload cores, and the cores would operate with the same or similar core power distribution core operating limits report (COLR) limits (e.g., AFD and rod insertion limits).

Reference:

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

RAI Question 5:

SQN is proposing a significant increase in boron concentration in the Refuel Water Storage Tank (RWST) and the cold leg accumulators.

a. *Please discuss the NRC approved methodology used to calculate the proposed boron concentrations for the RWST and the cold leg accumulators. Provide a reference to the NRC staff SER for this methodology.*

b. *The upper range of the proposed boron concentration in the RWST and the accumulators is 3800 ppm. Please discuss the technical basis for an upper limit on boron concentration for SQN, including the possibility of crystallation anywhere in the reactor coolant system and any associated auxiliary systems or equipment.*

c. *Please discuss the impacts of the increased RWST boron concentration on SQN refueling operation and procedures. Include in this discussion the impact on the likelihood and severity of a boron dilution event during refueling operations.*

Response:

a. The increase in boron concentration in the RWST and the cold leg accumulators is required to preclude re-criticality following a LOCA when the safety injection (SI) pumps are switched from the RWST to the sump for cold leg SI. The conceptual SQN TPC designs are more reactive at cold conditions than typical designs that do not have TPBARs. The increased reactivity results in an increase in the boron concentration required to preclude re-criticality following a LOCA. Furthermore, the cladding temperatures following a LOCA may cause the TPBARs to fail, reducing the amount of Li^6 in the core and further increasing the boron concentration required to ensure post-LOCA sub-criticality. To offset the increases in the post-LOCA core reactivity, the boron concentration in the RWST and the cold leg accumulators must be increased.

   Framatome-ANP evaluated the containment sump post-LOCA boron concentration for the SQN TPCs to ensure that sufficient boron exists in the sump to preclude re-criticality when the SI pumps are switched from the RWST to the sump for cold leg SI. This evaluation was performed using the increased boron concentration in the RWST and the cold leg accumulators. Framatome-ANP used the NRC-approved core simulator, NEMO (reference), to calculate the pre-LOCA reactor coolant system (RCS) boron concentration. The borated water in the RCS was then mixed with the other sources of water that feed into the sump during a LOCA to determine the post-LOCA sump boron concentration. To verify sub-criticality, the post-LOCA sump boron concentration was compared to the post-LOCA critical boron concentration, which is calculated using the NEMO code. Post-LOCA sub-criticality is defined as K effective (K_{eff}) less than 1.0 and assumes: 1) all control rods are out, 2) no xenon is present in the core, 3) the LOCA occurs at any time in the core life, 4) a conservative number of TPBARs failed during the LOCA and 5) the containment sump inventory temperature range is 50 degrees Fahrenheit ($^{\circ}$F) to 212°F. Additional calculations were also performed to consider the possibility of sump dilution at the time of hot-leg switchover. Assuming conservative failures of TPBARs and various adverse reactivity conditions, sub-criticality requirements for large break LOCA are satisfied with the increased RWST and cold leg accumulator boron concentrations.

b. The upper range of the proposed boron concentration (3800 parts per million [ppm]) has been established to provide an upper limit of an acceptable range for RWST target boron concentration. Establishment of a range instead of a set value provides operational flexibility in meeting RWST boron requirements. The minimum acceptable temperature associated with the maximum RWST boron concentration is near the freezing point. Therefore boron precipitation from solution is not credible at the minimum RWST temperature of 60°F and cold leg accumulator minimum of 70°F for the targeted boron concentrations.

c. During refueling operations, the reactor cavity, vessel, and to some degree the spent fuel pool are all interconnected and will potentially approach the RWST boron concentration, since the RWST is the primary source of water supplied to the refueling canal during refueling operations. However, the higher boron concentration is not driven by refueling criticality requirements. The post-LOCA shutdown margins as discussed in response RAI Question 5.a above, require the higher boron concentrations to ensure acceptable shutdown margin in a post-accident period. The net result (benefit) of the higher RWST boron concentrations on refueling operations is an increase in boron concentration above that required for refueling. The current TSs note that the spent fuel pit (SFP) nominally has 2000 ppm boron concentration; this value exceeds the 300 ppm boron required to maintain $k_{eff} \leq 0.95$ and the 700 ppm required to protect against the most severe fuel handling accident. The higher boron concentrations provide additional margin, lessen the impact of any boron dilution event, and is bounded by the existing analysis. Since the likelihood of a boron dilution event is influenced by low or non-borated supplies being introduced into a borated body of water, the likelihood of a boron dilution event is not changed by the increased RWST boron concentration. Therefore, the higher RWST boron concentrations do not adversely affect either refueling operations or the likelihood of boron dilution events.

Reference:

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

RAI Question 6:

On page 2-10 of the SQN Topical Report, the licensee states that, "the axial length and position, the number of TPBARs per cluster, and the TPBAR ^6Li loadings 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." Please discuss the administrative controls that are placed on the design such that safety limits are not exceeded, and the training that analysts receive for designing tritium producing cores.

Response:

Formal training for core designers and analysts relative to design and analysis of TPBAR cores was not conducted. However, the TPBAR designer (PNNL) conducted several technical meetings with TVA and the Framatome-ANP designers and analysts to familiarize them with the TPBAR design, their effect on core behavior, and available reference documents. Since the TPCs are analyzed using existing codes and methods approved by the NRC, no aspect of TPC fuel cycle design required special training; for example, the cores were designed and analyzed using the approved nuclear design simulator code package, CASMO3/NEMO (References 1 and 2). The nuclear design bases for TPCs are described in Section 2.4.3 of the SQN Topical Report BAW-10237, (Reference 3) (pages 2-7 and 2-8) and are the same as those currently specified in the SQN Updated Final Safety Analysis Report (UFSAR) for standard reload cores. Consequently, TPCs are designed to meet the same safety analysis criteria as standard reload cores. The only new safety analysis criteria added by the TPBAR cycles relates to the minimum and maximum tritium production per rod as described in Section 2.4.3 of BAW-10237, Revision 1 (pages 2-11 and 2-12); these criteria will be checked by the core designer each cycle to assure compliance.

The design of TPCs is analogous to the design of non-TPCs. Recent SQN fuel cycles have used a combination of fixed burnable absorbers (BPRAs) and gadolinia as an integral absorber. The TPC substitutes lithium for boron as the fixed burnable absorber. FRAMATOME-ANP has current experience with the design and licensing of fuel cycles that use a combination of BPRAs and gadolinia (e.g., Three Mile Island Unit 1, Crystal River Unit 3, Davis-Besse, and SQN Units 1 and 2). New employees are trained in approved methods with guidance from experienced tutors. Each implemented fuel cycle undergoes a reload safety evaluation to ensure all key safety analysis limits are met.

References:

1. STUDSVIK/NFA-89/3, "CASMO-3 - A Fuel Assembly Burnup Program," Studsvik AB, Nykoping, Sweden, November 1989.

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

3. BAW-10237, Revision 1, "Implementation and Utilization of Tritium Producing Burnable Absorber Rods (TPBARS) in Sequoyah Units 1 and 2," September 2001

RAI Question 7:

On page 2-12 of the SQN Topical Report - To determine that the amount of tritium produced per rod will remain within the allowable maximum and minimum values the licensee considered uncertainties in various parameters. Please discuss the methodology used to ensure these uncertainties are conservative and applied conservatively.

Response:

Tritium production is calculated for each TPBAR pin value. It is based on the assembly average value calculated by NEMO and the burnups of the eight-neighboring pins are used to determine the local TPBAR pin concentration. The error terms considered are predicted-to-measured assembly power variations, manufacturing tolerances (fuel enrichment, lattice pitch, Gadolinia concentration, and TPBAR concentration), local tritium uncertainties, Cycle N-1 shutdown flexibility, and power level uncertainty. These are conservatively combined to obtain a total uncertainty of 13.3%. A factor of 1.133 will be applied to the maximum predicted tritium and a factor of 1.133 will be divided into the minimum predicted tritium produced in a pin before a comparison to the maximum and minimum limits is performed. The predicted tritium production for the WBN LTAs for 32 pins was compared to the measured values. All the predicted tritium concentrations were within ±5% of the measured concentrations with a mean of -1.2% and a standard deviation of 1.5% (measured accuracy is estimated as ±4%). The negative mean implies an under-prediction by NEMO. As more data is accumulated, the tritium uncertainty will be revised using a one-sided 95/95 tolerance/confidence limit of the tritium measured to predicted variations. As available, as-built lithium loadings for a particular fuel cycle may be used to reduce the tritium production uncertainty associated with the manufacturing tolerance.

RAI Question 8:

On page 2-13 of the topical report, the licensee states that conservative augmentation factors were defined and applied to the limiting power peaking factors when peaking margins were calculated. These augmentation factors were applied to account for the effects of flux peaking caused by axial gaps between absorber pellets in a pellet stack or between pellets in adjacent pencils. NUREG-1672 established a nuclear requirement that gaps between pellets shall cause power peaking of less than 3 percent for burnups less than 10,000 MWD/MTU and less than 5 percent for burnups above 10,000 MWD/MTU. Please discuss how these augmentation factors were calculated and applied, and the power peaking margins available.

Response:

The nuclear requirement for peaking caused by gaps between pellets stated in NUREG-1672 was revised several times by the TPBAR designer (PNNL) subsequent to the review of the TPC topical report. PNNL provided instructions for Framatome-ANP to use for analysis of the SQN TPCs in PNNL Document TTQP-1-116 (the values in Reference 1 were used in the analysis for Section 2.4.3 of BAW-10237, Revision 1). TTQP-1-116 discusses how the factors were calculated by PNNL and provides illustrations of the factors.

In the power distribution analysis, the increase in peaking due to the gaps defined in TTQP-1-116 were applied as peaking augmentation factors when peaking margins were calculated (i.e., the peaks calculated by the nuclear design simulator code NEMO [Reference 2] were increased by the gap factors to account for the effect of the gaps on power peaking). The gap peaking augmentation factors were applied as a function of fuel enrichment, ^{6}Li concentration, TPBAR pattern type, location of the fuel rod within the assembly (i.e., fuel rod location relative to the TPBAR rodlets), and elevation along the fuel rod as prescribed by TTQP-1-116. These factors were bounding for all burnup values.

For application to DNB peaking margins, Framatome-ANP determined the effect of the gap peaking factors specified in TTQP-1-116 on DNBR (see response to Question 1), and DNB MAP reduction factors were defined for use in the core power distribution evaluation. The MAP reduction factors replaced direct application of the gap peaking factors from TTQP-1-116 for calculation of DNB peaking margins.

Based on the results of the peaking margin calculations, it was observed that the TPBARs depress the power in adjacent fuel rods so that peaking factors in fuel rods affected by gaps tend to be either non-limiting or no more limiting than those in standard reload core designs (see the response to Question 4 for more details on available peaking margins). Therefore, peaking increases caused by gaps between the TPBAR pencils can be accommodated by the 96-feed TPBAR cores evaluated for SQN.

For completeness, it is noted that PNNL modified the gap factors slightly after the release of TTQP-1-116 (the current revision level of the document is Revision 9). Framatome-ANP evaluated the revisions to the gap peaking factors and verified that the modifications did not adversely impact the results presented in Section 2.4.3 of BAW-10237.

References:

1. PNNL Document TTQP-1-116, Revision 4, "Production TPBAR Inputs for Core Designers," April 21, 2000.

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

RAI Question 9:

To accommodate TPBARs, the licensee determined that four rod cluster control assemblies must be relocated in order to ensure shutdown margin requirements are satisfied. Please provide the technical basis for this proposed change, including a discussion of the analyses performed in support of this proposed modification and the NRC-approved methods used to perform these analyses. How does this modification impact the results of the Updated Final Safety Analysis Report, Chapter 15, transient analyses? Does the licensee plan to submit this proposed modification to the NRC for review and approval as part of a separate license amendment request package?

Response:

The rod cluster control assembly (RCCA) relocation discussed on page 2-10 of Framatome-ANP Topical Report No. BAW-10237 was designed to provide the maximum increase in shutdown margin through the use of existing installed plant equipment. As part of the SQN original nuclear steam supply system scope of supply, spare control rod drive mechanisms (CRDMs) were installed over core locations E-5, E-11, L-5, and L-11 during initial plant construction. The modification takes advantage of the spare CRDMs (and their central core location) by transferring RCCAs from periphery core locations D-2, P-4, B-12, and M-14 to the spare CRDM locations. The new location of the RCCAs increases the overall shutdown margin provided by the rod control system without adding additional RCCAs or modification of the present rod control system.

To ensure the availability of the spare CRDMs to support the TPC, the RCCA relocation was implemented on SQN Units 1 and 2 during the Cycle 11 refueling outages (September 2001 for Unit 1 and April 2002 for Unit 2). The modifications were performed under the plant design change process and were evaluated prior to implementation in accordance with the requirements of 10CFR50.59. The evaluations addressed the mechanical and electrical changes required to activate the CRDMs at core locations E-5, E-11, L-5, and L-11 (and designate the CRDMs at core locations D-2, P-4, B-12, and M-14 as spare components) as well as the functional changes associated with the revised RCCA core locations. The specific evaluations performed for the subject modification include:

1. Mechanical Changes - Cover plates installed on the upper end of the control rod guide tubes in the reactor vessel upper internals assembly were removed from core locations E-5, E-11, L-5, and L-11. New cover plates were installed on the upper end of the control rod guide tubes for core locations D-2, P-4, B-12, and M-14. These changes were evaluated for their effect on component integrity and coolant flow characteristics. The evaluation concluded that the new cover plates were consistent with the material and structural design requirements of the SQN reactor vessel internals. The cover plate modifications have a minimal effect on the flow of reactor coolant in the upper internals and do not alter core bypass flow.

2. Electrical Changes - The spare CRDMs and control rod drive position indication instrumentation for core locations E-5, E-11, L-5, and L-11 are identical to the mechanisms and instrumentation installed at core locations D-2, P-4, B-12, and M-14. The required power, instrumentation and control cables feed the CRDMs through a connection panel located on the reactor vessel head assembly. The cables terminate with removable connectors at the panel to allow for routine cable removal and reconnection during removal of the reactor vessel head assembly for refueling. As part of this modification, the power, instrumentation and control cables connected the CRDMs at core locations D-2, P-4, B-12, and M-14 were reconnected to the CRDMs at core locations E-5, E-11, L-5, and L-11 at the connection panel. The CRDMs in core locations D-2, P-4, B-12, and M-14 originally functioned as Shutdown Bank A, Group 1 mechanisms (refer to Section 7.7.1.2.1 of the SQN UFSAR for a general description of rod control system operation). Following the modification, the CRDMs at core locations E-5, E-11, L-5, and L-11 function as Shutdown Bank A, Group 1 mechanisms. Because the CRDMs and position

indication instruments were identical, redirection of the power, instrumentation and control cables did not have any affect on the control rod drive control system or the rod position indication system.

3. Core Alterations - Handling and positioning of the RCCAs and thimble plugs required to permit connection of the RCCAs and drive rods were performed during routine refueling operations using existing plant procedures. No changes to the RCCAs, drive shafts or thimble plugs were required to support the subject modification.

4. Functional Changes - Functional changes associated with the RCCA relocation were integrated with the normal Cycle 12 core designs and reload safety evaluations of each SQN unit. Framatome-ANP evaluated the SQN Cycle 12 reload cycle designs using methodology developed specifically for analyzing the operation and safety of Westinghouse-designed PWRs. References 1 through 4 describe this methodology, which has been reviewed and approved by the NRC. References 5 and 6 document the license change request submitted by TVA to the NRC to incorporate the Framatome-ANP Mark-BW fuel in SQN. The NRC approved the change request in Reference 7. The RCCA relocation does not result in a departure from the method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis, therefore, no changes to the mentioned NRC approved methods are required.

The Framatome-ANP evaluation of the SQN Cycle 12 Units 1 and 2 reloads determined that the SQN Cycle 12 core designs met all applicable design criteria and all pertinent licensing basis acceptance criteria. Operating limits, safety limits, and all core related safety parameters have been considered. The RCCA relocation does not increase the frequency of occurrence or the consequences of any previously evaluated accident in the SQN UFSAR, nor does the RCCA relocation create the possibility for an accident of a different type than any previously evaluated in the UFSAR. Also, the RCCA relocation does not increase the probability of occurrence or the consequences of a malfunction of a system, structure, or component (SSC) important to safety that has been previously evaluated in the SQN UFSAR, nor does the RCCA relocation create the possibility for the malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. Finally, the RCCA relocation will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. Therefore, the relocation of the four RCCAs does not have any negative impacts on the UFSAR Chapter 15 transient analysis.

References:

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

2. BAW-10169P-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," Babcock & Wilcox, Lynchburg, Virginia, October 1989.

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

4. BAW-10168P-A, Revision 3, "RSG LOCA - BWNT Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," B&W Nuclear Technologies, Lynchburg, Virginia, December 1996.

5. Letter from R. H. Shell (TVA) to NRC Document Control Desk, TVA-SQN-TS-96-01, April 4, 1996.

6. Letter from R. H. Shell (TVA) to NRC Document Control Desk, TVA-SQN-TS-96-01, Revision 1, February 7, 1997.

7. Safety Evaluation by the Office of the Nuclear Reactor Regulation Related to Amendment 223 to Facility Operating License No. DPR-77 and Amendment 214 to Facility License No. DPR-79, April 21, 1997.

RAI Question 10:

For an extended shutdown near End of Life the buildup of ^{3}He through tritium decay can have a significant impact on core reactivity. On pg. 2-16 of the SQN Topical Report the licensee states that the reactivity effects of an extended shutdown will be evaluated for each reload cycle in the cycle-specific reload safety evaluation and that 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 following an extended shutdown. Please discuss the type of guidance which will be provided to the analyst and how each of these requirements will be administratively controlled.

Response:

Framatome-ANP will provide guidance for use by the reactor operators and engineers to identify and assess parameters that would indicate that a formal evaluation of an extended shutdown should be conducted. Those parameters could include time in cycle burnup when the shutdown occurs, the anticipated length of unit down time, and operational history leading up to the extended shutdown. The focus of the guidance would be to validate the cycle-specific safety analysis checks and to ensure that the core reactivity and power distribution limits specified in the TSs and COLR would remain valid, or be revised (as appropriate), prior to resumption of power operation.

Similarly, Framatome-ANP will evaluate the effects of an extended shutdown on the data in the operational data package for each cycle with TPBARs. The limitations on the applicability of the data will be included with the operational data package on a cycle-specific basis. In the event of an extended shutdown, Framatome-ANP would be contacted before the data becomes invalid and the data would be reassessed as necessary to support resumption of power operation. The existing administrative controls governing the use of the operational data package will ensure that the data used is applicable.

RAI Question 11:

In Section 2.4.4 of the SQN Topical Report the licensee states that the BWCMV-A and the BWU Critical Heat Flux correlations were utilized in performing DNBR analyses. Please provide technical justification regarding the applicability of these correlations for Babcock & Wilcox 17x17 fuel with the production TPBARs designed for SQN.

Response:

The technical justifications for the application of the BWCMV-A and BWU critical heat flux (CHF) correlations to Mark-BW17 fuel are provided, respectively, in References 1 and 2. The applicability of these two correlations to the Mark-BW17 fuel design is not impacted by presence of the TPBARs. The fuel geometry of the Mark-BW17 is compatible with the TPBAR assembly and does not require any modification to accommodate the new absorber assembly. Furthermore, the correlations continue to be applied within the range of limitations identified in Table 1 of the SER for each of the correlations. This results in BWU being used for the low pressure conditions of the steamline break analysis, while BWCMV-A is used for all other DNB analyses.

The effect of the TPBARs on local coolant conditions have been addressed in the analyses which support the SQN Topical Report. As discussed in the response to Question 1, the demonstration analyses which support the topical report determined that the impact of the TPBARs on the subchannel flow is adequately addressed by applying a bounding bypass assumption. Furthermore, the global effects of the TPBARs on the fuel rod powers and axial power shapes are addressed through Framatome-ANP's standard reload methods which establish core operating and safety limits by evaluating the cycle specific core power distribution. Finally, the effects of the local power peaking perturbations that could occur due to axial gaps within the TPBAR absorber stack are accounted for in the core power distribution analysis, through the use of DNB based MAP reduction factors, and, in the steamline break analysis, through a direct DNBR penalty.

References:

1. BAW-10189P-A "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," Framatome Cogema Fuels, January 1996.

2. BAW-10199P-A "The BWU Critical Heat Flux Correlations," Framatome Cogema Fuels, August 1996.

RAI Question 12:

The licensee developed a 24-channel LYNXT model to evaluate the local coolant and surface temperature conditions within the thimble tubes occupied by TPBARs. Please provide a discussion and the technical basis for the lateral crossflow resistance factors applied between the thimble tube channels and surrounding channels.

Response:

The guide tube side flow holes are modeled explicitly in LYNXT using the combination of axial node length and crossflow gap width. The flow area of the crossflow gap is set to match the actual area of the guide tube flow hole. The crossflow resistance for the guide tube flow hole is based on a sharp edged orifice in a wall with infinite surface area. The lateral crossflow resistance factor of 2.8 is extracted from the reference.

Reference:

"I.E. Idelchik, Handbook of Hydraulic Resistance, Hemisphere Publishing Corporation, Washington, D.C., 1986."

RAI Question 13:

In Section 2.15.2 of the SQN Topical Report the licensee states that all non-LOCA key safety analysis parameters for a core with TPBARs remain bounded by the parameters used in the current applicable safety analysis for SQN. The licensee does not provide any discussion regarding the magnitude of the impact that the TPBARs have on these key safety analysis parameters. Please discuss the impact of the TPBARs on the margin remaining to the assumed key safety analysis parameters. Include in this discussion the impacts of the change in most negative Doppler-only power coefficient at hot zero power conditions (discussed on page 2-14 of the SQN Topical Report).

Response:

The table below compares the limit value of the key safety analysis parameters to the cycle specific values from a 96-feed equilibrium SQNTPC cycle, a 96-feed first transition SQNTPC cycle, and a recent SQN cycle without TPBARs (SQN Unit 2 Cycle 12). The design with no TPBARs differs from the SQNTPC designs in the feed enrichment, the feed batch size, the loading pattern, and the distribution and isotopic concentrations of burnable poisons, gadolinia, and TPBARs. Likewise, the SQNTPC designs differ in feed enrichment, loading pattern, and the distribution and isotopic concentrations of burnable poison, gadolinia, and TPBARs. All of these differences in the cycle designs have an impact on the key safety analysis parameters and are modeled in the NEMO core simulator for the SQN reload safety evaluations.

The comparison in the table shows that all three designs fall within the limits and ranges of the kinetics parameters assumed in the safety analysis with the exception of the most negative Doppler-Only Power Coefficient (DOPC) at hot zero power (HZP). As described on page 2-14 of the of the SQN Topical Report and in the footnote to the table below, instances where the cycle specific value of the most negative DOPC exceeds the -19.4 pcm/% full power limit near zero power are acceptable. The table also illustrates that the SQNTPC designs have comparable margin to the core design that does not have TPBARs.
*

Comparison of Core Kinetics Parameters

Parameter	Safety Analysis Limit Value	96-Feed Equilibrium TPBAR Cycle	96-Feed 1st Transition TPBAR Cycle	Sequoyah Unit 2 Cycle 12 (No TPBARs)
Moderator Coefficient, pcm/F HZP, Maximum Hot Full Power (HFP), Maximum All, Minimum	 < +7* < 0 > -45	 -2.66 -10.68 -35.73	 -1.21 -9.17 -35.86	 -0.44 -7.73 -36.34
DOPC, pcm/%FP Least Negative, HZP Least Negative, 100%FP Most Negative, HZP Most Negative, 100%FP	 < -10.2 < -6.5 > -19.4 > -12.5	 -14.16 -6.55 -20.25** -10.26	 -14.20 -6.71 -21.01** -10.53	 -11.60 -7.05 -21.42** -10.73
B_{eff}, % Beginning of Life (BOL), HFP HZP	$0.55<B_{eff}<0.75$ $0.55<B_{eff}<0.75$	0.67 $0.66\leq B_{eff}\leq 0.68$	0.66 $0.66\leq B_{eff}\leq 0.67$	$0.64\leq B_{eff}\leq 0.65$ $0.63\leq B_{eff}\leq 0.64$
B_{eff}, % End of Life, HFP HZP	 $0.44<B_{eff}<0.75$ $0.45<B_{eff}<0.75$	 0.56 0.56	 0.55 $0.55\leq B_{eff}\leq 0.56$	 0.53 $0.53\leq B_{eff}\leq 0.54$
Minimal Trippable Worth, pcm	> 4000	6167	5714	5455
Minimum Shutdown Margin, pcm	≥ 1600	2379	1926	2027
Refueling K_{eff}	< 0.95	<0.95	<0.95	<0.95

* TSs currently require a moderator coefficient <0 pcm/F. Rod withdrawal limits will be used, if required, to ensure HZP MTC remains <0 pcm/F.

** At zero power and end of cycle, flux redistribution causes the DOPC to be more negative than the limit. Note that this is not specific to TPBAR core designs. 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 a 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 specific values of the most negative DOPC exceeding the -19.4 pcm/%FP limit near zero power is acceptable.

RAI Question 14:

In Section 2.15.5.1 of the topical report the licensee states that, "there are instances when the thimble/TPBAR can be heated, rather than cooled by the fluid in the surrounding channels."

Please discuss these conditions and the expected increase in TPBAR temperatures. Why are these temperatures acceptable?

Response:

Because of the low power produced by the TPBARs (7.80 kW/rod per Table 3.3-3 of the reference) during both the large and small break LOCA there are periods where the thermal response of the TPBAR lags behind the hot channel fluid temperature. Under these conditions, the surrounding fuel rods govern the channel fluid conditions with the guide thimble/TPBAR serving as a heat sink. This was taken into account in the analysis cases performed for the TPBARs under LOCA conditions. When this condition persisted, the analysis assumed an artificially high surface heat transfer coefficient to effectively heat the TPBAR via forced convection in addition to its internal nuclear heat generation and radiation from the surrounding fuel rods. When an equilibrium temperature was obtained or the fluid temperature became less than the thimble/TBPAR temperature, convection to the fluid channel was then neglected. This effect is quantified in Section 2.15.5.1 of the reference as the "Upper Bound" cases.

In these situations, as shown in the report, the thimble/TPBAR temperature remained below the 10CFR50.46 criteria of 2200°F. Thus, this temperature is acceptable for the thimble. The TPBAR pressure boundary is made of 316 stainless steel. This material is not prone to the metal/water reaction like zirconium at these temperatures and as such will not be affected.

Reference:

BAW-10237, Revision 1, "Implementation and Utilization of Tritium Producing Burnable Absorber Rods (TPBARS) in Sequoyah Units 1 and 2," September 2001.

RAI Question 15:

With respect to calculation of TPBAR temperatures, assumptions 2.15.5.1.5 and 2.15.5.1.7 of the SQN Topical Report include the following two statements which are not clear and need to be better defined, ". . . lack of significant steam flow" and ". . . low heatup rates." Please provide a more detailed quantitative discussion of what is meant by these two statements, including technical justification for these assumptions.

Response:

With regard to the first statement, "lack of significant steam flow," the report was establishing that vapor flow in the annulus between the TPBAR and thimble is small under LOCA conditions when compared to that experienced in the channels between the fuel rods. The statement was made to defend the modeling of the thimble exterior surface with regard to the metal/water reaction. This is appropriate given:

1) the surface heat flux on the TPBAR is much lower (by more than an order of magnitude) than the surrounding fuel rods; thus vapor generation within the thimble will be lower than on the outside, and

2) the dashpot openings at the thimble bottom have a very small flow area which limit the amount of water/vapor entering the thimble.

These factors limit interior transient oxidation. However, if these are totally neglected, local oxidation will still remain acceptable. The upper bound TPBAR analysis case has a maximum exterior clad oxidation of 7.95%. Assuming this also occurs on the inside of the thimble, this results in a total oxide thickness of 15.9%, which is less than the 17% limit.

With regard to the second statement, "low heatup rates," analyses have shown that the TPBAR heat-up rate is approximately 10°F/sec under LOCA conditions. The low heat-up rates, coupled with the low mass of the material (6.2 gram [gm]/centimeter [cm]) and the thermal conductivities of the materials within the TPBAR assembly (stainless steel, zirconium, and lithium aluminate) allow the temperature gradients within the TPBAR to be small. A quantitative value of this is not available since the TPBAR itself was modeled as a lumped mass in LOCTAJR. However, if the distinct volumes were to be modeled, it will not have a significant impact on the outcome since, in the extreme upper bound case, the difference between the guide thimble temperature and the TPBAR is less than 20°F.

RAI Question 16:

In Section 2.15.5.1 of the SQN Topical Report the licensee states that the boundary conditions (fuel rod temperatures and fluid conditions) for the TPBAR temperature calculations are taken from the Appendix K LOCA analyses of record. Modeling of the downcomer region and downcomer boiling have recently been shown to substantially impact peak clad temperature (PCT) and oxidation following a loss-of-coolant accident (LOCA), especially for ice condenser containments. Please discuss how the downcomer region and downcomer boiling are modeled in the SQN LOCA Appendix K evaluation model, and dicuss any potential adverse impacts this modeling may have on PCT, oxidation, and TPBAR temperatures and oxidation.

Response:

Framatome-ANP uses an NRC-approved Appendix K evaluation model as the licensing basis for fuel operational limits at SQN. The "worst" case LOCA transient, in terms of PCT response, was employed to establish boundary conditions for predicting post-LOCA TPBAR thermal response. The level of detail in the current evaluation model is insufficient to address the reactor vessel downcomer boiling phenomena alluded to by the NRC in this question.

Framatome-ANP is aware of the potential effect that post-LOCA downcomer boiling might have and is working with the NRC to better define and resolve the issue. Framatome-ANP believes that insufficient analysis has been conducted to determine whether the degree of post-LOCA downcomer boiling would have a significant adverse effect on the peak cladding temperature and cladding oxidation for SQN. In addition, there is no current substantive basis for defining the effects of downcomer boiling on post-LOCA TPBAR thermal response.

Framatome-ANP discussions with the NRC regarding downcomer boiling were initiated under a separate contract. It is expected that the outcome of those discussions will define a process and schedule for resolution of this issue. Subsequent to resolution, an evaluation of downcomer boiling on the TPBAR post-LOCA thermal response will be warranted. Framatome-ANP believes that a high degree of conservatism in PCT, cladding oxidation, and post-LOCA TPBAR thermal response predicated on its Appendix K LOCA methodology will ultimately be demonstrated.

RAI Question 17:

Please provide references to the approved LOCA analysis methodologies applied for SQN. Also provide a statement that SQN and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

Response:

Current NRC-approved computer codes and evaluation methodologies that serve as the SQN LOCA licensing basis are described in depth in the SQN reload fuel topical (Reference 1). Relevant LOCA methodology is described in Reference 2.

Applicability of the TVA-supplied inputs to the SQN LOCA analyses is confirmed by TVA each fuel cycle. The impact of changes to plant component or operational configurations on LOCA analyses of record are addressed as necessary as part of the plant change process. Changes to fuel assembly design or materials and their impact on existing LOCA calculations are addressed each fuel cycle to assure that current analysis results remain applicable and bounding and relevant acceptance criteria are met for current fuel configurations.

References:

1. BAW-10220P-A, "Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2," Framatome ANP, Inc., November 2001.

2. BAW-10168P-A, Revision 3, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Framatome ANP, Inc., B&W Nuclear Technologies, Lynchburg, Virginia, December 1996.

RAI Question 18:

Please provide a complete description of the boric acid accumulation evaluation model that is used to establish compliance with Title 10, Code of Federal Regulations, Section 50.46(b)(5) and provide a complete assessment of model conservatisms and non-conservatisms. In addition, please compare your evaluation model prediction to your procedures for initiating hot-leg injection and assess conservatisms and non-conservatisms associated with the procedures.

Response:

See Attachments 1 and 2 for the Proprietary and Non-Proprietary responses respectively.

RAI Question 19:

Section 2.15.6.5 of the SQN Topical Report discusses the Steam Generator Tube Failure event. The licensee states that a conservative analysis of the potential offsite doses resulting from this accident is presented, including an updated thermal and hydraulic analysis, and that this analysis incorporates conservatively updated assumptions. Please provide a discussion of the updated thermal and hydraulic analysis that was performed to address TPBARs, including a comparison of the updated to the previous assumptions, and the updated sequence of events. Also, please provide the basis for assuming two TPBARs fail.

Response:

The analysis of record (AOR) for the SQN Steam Generator Tube Rupture (SGTR) uses a transient code to calculate the initial plant response to the SGTR until equilibrium is reached between the break flow and the SI flow. The calculation considered break flow continuing for 30 minutes before being terminated as a result of appropriate operator actions. The analysis did not assume any other specific operator actions.

The updated SGTR analysis maintains the same basic assumption that break flow is terminated 30 minutes after the tube rupture, but uses a more conservative break flow model (refer to Table 19-1 for a comparison of the total break flow in the updated analysis to the AOR). The updated analysis models a lower main steam safety valve pressure to account for setpoint tolerance and blowdown. This contributes to higher post-trip and equilibrium break flows. The AOR assumed a secondary pressure of 1100 psia after trip, while the updated analysis models the secondary pressure of 940.4 psia. The updated analysis used SQN specific maximum SI flow data which are consistently higher than the assumed values used in the AOR for the pressure range applicable to the analysis. This also contributes to higher equilibrium break flow.

The AOR did not include consideration of flashing of break flow as it enters the ruptured steam generator (SG). The updated analysis conservatively calculated the flashing fraction assuming that all break flow is at the hot leg temperature. Since the analysis does not model the operator actions that would lead to break flow termination, which includes cooling of the RCS to below the saturation temperature corresponding to the ruptured SG pressure, applying this flashing fraction for the duration of the break flow is a significant conservatism relative to the expected transient response.

The AOR did not include detailed modeling of pre-trip and post-trip break flow or steam releases. The higher break flow in the updated analysis results in earlier reactor trip and SI actuation. Earlier reactor trip is conservative since the post-trip releases are discharged directly to the atmosphere via the main steam safety valves.

The updated analysis conservatively bounds cases with and without tube plugging, for the current and replacement SGs, and includes the impact of the 1.3% main feedwater leading edge flow meter power uprate.

Table 19-1 compares the key thermal and hydraulic results of the AOR and the updated analysis. Table 19-2 shows the sequence of events for the updated analysis.

The assumption of two TPBAR failures was made to include conservatism in the radiological consequences. It should be noted that TPBARs do not experience damage as a result of the SGTR; rather, the two TPBARs are assumed to have failed during normal operation prior to the event. This is an extremely unlikely occurrence.

Table 19-1
Comparison of AOR and Updated Analysis

	AOR	Updated Analysis
Total Tube Rupture Break Flow (0 - 30 minutes)	131,250 pound mass (lbm)	172,700 lbm
Percentage of Break Flow that Flashes	No Flashing Modeled	Pre-trip: 18.0% Post-trip: 4.74%
Ruptured SG Steam Releases: Pre-trip Post-trip until 30 minutes	 Not Modeled 46,800 lbm	 78,100 lbm 70,100 lbm
Intact SG Steam Releases Pre-trip Post-trip until 2 hours 2 hours until 8 hours	 Not Modeled 429,000 lbm 1,080,000 lbm	 232,000 lbm 530,000 lbm 1,237,000 lbm

Table 19-2
Updated Analysis Sequence of Events

Event	Time
Tube Rupture Occurs	0 sec
Reactor Trip	65 sec
Safety Injection Initiated	73 sec
Auxiliary Feedwater Initiated	133 sec
Break Flow and Ruptured Steam Generator Steam Releases Terminated	30 minutes
Intact Steam Generators' Steam Releases Terminated	8 hours

RAI Question 20:

Regarding the thermal-hydraulic evaluation of the TPBARs discussed in Section 3.6:

a. *Please provide a listing of the NRC-approved analytical codes and methods used to evaluate the bypass flow and thermal performance of the TPBARs.*

b. *Please quantify the margins remaining for thermal hydraulic acceptance criteria.*

c. *Please discuss any uncertainty considered in these evaluations and provide justification for not applying additional uncertainties to power, temperature and pressure, which are* *assumed to be at nominal conditions. This is of particular interest for the no bulk boiling requirement which appears to have very little margin.*

d. Please provide a profile for the bounding axial power shape that was used for these analyses and discuss how it was conservatively selected.

Response:

Section 3.6 discusses three aspects of the thermal-hydraulic design: bypass flow, guide tube boiling, and TPBAR component temperatures. Each of these will be addressed separately.

<u>Bypass Flow</u>

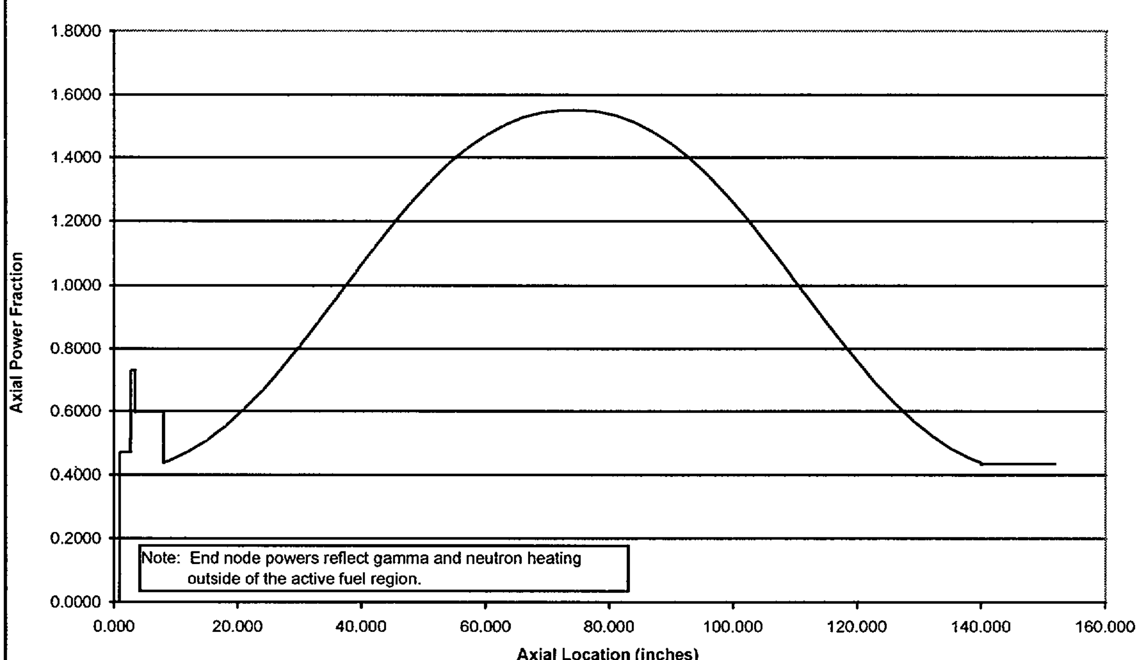
Bypass flow in the TPBAR core was determined by first establishing a one dimensional, steady state flow network which modeled all of the bypass flow paths in the SQN core. Next, various configurations of TPBARs and control components were modeled. By solving the one-dimensional form of the mass, momentum and energy equations for flow rate and momentum, elevation, friction and form loss pressure drops, the total bypass flow rate for the various core configurations could be determined. The sensitivity analysis showed that the various core configurations, both with and without TPBARs, exhibited guide tube bypass fractions that varied by less than 0.5%, and in no case, did the total core bypass exceed the 7.5% design value already in place for SQN.

<u>Guide Tube Boiling</u>

The calculations for guide tube boiling, both bulk boiling and surface boiling in the dashpot, were performed with a 24-channel LYNXT model (Reference 1). Consistent with analyses documented in the original TPC Topical Report (Reference 2), the analysis of record for guide tube boiling did not consider uncertainties on core power, temperature and pressure, given the conservatism included in some of the other input parameters. Specifically, the limiting design analysis considered a 15% uncertainty on the gamma and neutron heating within the guide tube (which included the guide tube structure, the water within the guide tube, and the TPBAR). In addition, the limiting design analysis considered the minimum thermal design flow rate (including uncertainty), with the bounding 7.5% design bypass value. Finally, the analysis of record considered a limiting pin power distribution, where the four rods face adjacent to the guide tube containing the TPBAR were set at the 1.70 $F_{\Delta H}$ design limit, and the four rods diagonally adjacent were set at 1.62. (The $F_{\Delta H}$ of the limiting hot pin in a typical TPBAR fuel cycle is approximately 1.40.) Therefore, there is considerable conservatism applied to the power peaking and heat generation within the guide tube containing the TPBAR and the surrounding channels. The axial power shape used in the analysis of record was a symmetric axial shape with a 1.55 axial peak (see Figure 20-1). A scoping study was performed to evaluate inlet, outlet, and symmetric axial shapes of different magnitudes. From that study it was determined that, since guide tube boiling is a long-term phenomenon, a symmetric axial shape with a 1.55 magnitude peak conservatively represents long-term, steady-state operation.

The maximum bulk coolant temperature in the guide thimble containing the TPBAR was shown to be 651.4°F, which indicates >1°F margin to the saturation temperature at 2250 psia, with a 0.8% margin in local quality. The maximum cladding surface temperature was shown to be 654.4°F, against a design value of 662.4°F, while the clad surface temperatures within the dashpot region were less than 600°F, well below the temperature range where surface boiling would occur.

Figure 20-1
Axial Power Shape for Guide Tube Boiling



TPBAR Component Temperatures

Each TPBAR resides in the guide thimble within the fuel assembly and is cooled by reactor coolant that flows up the annulus between the TPBAR and the guide thimble tube. Heat is generated in the TPBAR from two sources: 1) the $^{6}Li(n,\alpha)^{3}H$ reaction in the absorber pellets, which produces one triton, one helium atom, and 4.8 million electron volts (MeV) of energy per reaction; and 2) gamma heating in the TPBAR components. The coolant in the annulus is heated slightly by the TPBAR, but gains more heat due to heat transfer from the coolant on the outside of the guide thimble. The LYNXT model provided the temperatures of the guide thimble inlet and outlet coolant temperatures that are used as the boundary conditions for the TPBAR external cladding surface with a linear distribution between the top and bottom of the TPBAR. The temperatures of the internal TPBAR components were evaluated from the heat inputs predicted from the n-α reaction and the gamma heating using one-dimensional heat transfer calculations.

Both normal operating heat loads and 118.7% overpower (Condition II at BOL) were used as input to determine component temperatures. These melting limits provide significant margins above operating temperatures as tabulated in the following:

Parameter	Melting Limit
Cladding Temperature (°F)	2568.9
Getter Temperature (°F)	1759.7
Pellet Temperature (°F)	3199.7
Liner Temperature (°F)	1759.7

References:

1. BAW-10156-A, Revision 1, "LYNXT - Core Transient Thermal-Hydraulic Program, Revision 1," August 1993.

2. NPD-98-181, Revision 1, "Tritium Production Core (TPC) Topical Report," February, 1999.

RAI Question 21:

Section 3.7 of the NRC Staff SER (NUREG-1672) for the TPCTR states that "The higher reactivity worth of the lithium-6 in the TPC [tritium production core] relative to boron-10 used to control core reactivity, and the current experience base in producing lithium-6 enriched aluminate, impose a tight lithium-6 loading tolerance of 0.030 g/inch ±4.2 percent (±0.00125 g/inch) on an individual pencil basis." Section 3.7 of the SQN Topical Report revises this to a range of 0.028 to 0.040 ±0.00125 g/inch. Please provide the technical justification for this change, including the methods used to assess the change and the impacts on core reactivity.

Response:

PNNL has justified the use of lithium concentrations between 0.028 and 0.040 grams/inch in TPBARs for use at SQN (Reference 1). The selection of the lithium concentration should be as large as possible to provide adequate tritium production while remaining below the maximum TPBAR tritium production criterion determined by PNNL; however, the use of multiple concentrations in a core can be used for power distribution control, similar to the manner in which Framatome-ANP currently uses different boron loadings among BPRAs in core designs. The lithium content in the TPC is complemented with the integral gadolinia absorber. Thus the level of reactivity control in the TPC is accomplished with a combination of the lithium and gadolinia. Changes in the lithium loading due to either the lithium concentration or the number of TPBARs can be compensated by the change in the gadolinia loading. The lithium concentrations are explicitly modeled by the NEMO core simulator; therefore, their effects on power distribution and core reactivity are inherently considered in each phase of the reload safety evaluation.

Reference:

TTQP-1-116, Revision 9, "Production TPBAR Inputs for Core Designers," Pacific Northwest National Laboratory, Richland, Washington, April 2001.

RAI Question 22:

Section 3.7.3 of the SQN Topical Report includes a discussion of operation with catastrophic TPBAR failure. Please provide an outline of the types of decisions the operators will need to consider in order to ensure that power operation could continue without adverse consequences to fuel design and safety limits.

Response:

If a catastrophic TPBAR failure should occur, the effects of the failure will be evaluated by the existing Fuel Integrity Assessment Team (FIAT). The FIAT, which consists of representatives from Operations, Chemistry, Reactor Engineering, and Nuclear Fuels, will evaluate information from flux maps and identify compensatory measures required by the TSs. If the failure occurs at a time in life when adequate peaking factor margin is available, then no compensatory actions will be required. If, however, peaking factors exceed limits, then appropriate compensatory measures per TSs will be implemented to ensure that normal operation peaking factor limits will be met, assuming a single TPBAR failure. The compensatory measures could include:

- reduction in reactor operating space (e.g., modified AFD Limits)
- power derating

Initial compensatory measures will be defined within 5 days of detection of the TPBAR failure. The compensatory measures will be adjusted as operation proceeds based upon periodic flux map peaking factor determinations.

RAI Question 23:

Table 4-1 of the SQN Topical Report, Section 5.4.7 summarizes the plant specific evaluation performed to determine the net effect of TPC on residual heat removal (RHR) System cooling capability.

a. *Did this analysis consider the increased heat load from the spent fuel pool cooling system as a result of TPBARs being stored in the spent fuel pool?*

b. *Please quantify the impacts of a TPC on the time required for the RHR system to cool the reactor coolant system assuming both two-train (normal) and single-train cooldowns.*

Response:

The increase in heat load imposed on the SFP as a result of TPC has been quantified to be less than 0.5 MWt and includes both the increase in reactor core decay energy and higher decay energy in the SFP from the cumulative affect of multiple TPC core discharges over the time to full capacity of the SFP. The heat load imposed on the SFP from the individual TPBARS at 3 watts per TPBAR (less than 0.007 MWt total) is insignificant relative to the overall heat load in the SFP. This small increase in heat load has an insignificant impact on the Component Cooling System (CCS) which cools both the SFP heat exchangers and the RHR heat exchangers. The RHR heat load at the start of RHR cooldown is approximately 38 MWt (at time = 4 hours after shutdown). Existing analysis for thermal energy removal from the plant via the SFP CCS, RHR, CCS, and essential raw cooling water (ERCW) systems for all operating modes have been revised as necessary to account for the small change in heat loads imposed by the earlier offloads of fuel to the SFP, which completely bound any impacts resulting from TPC operation.

The Westinghouse RHR plant cooldown analyses uses the ANS-5.1-1979 decay heat standard which is based on design basis reactor power. Reactor power is not changing as a result of TPC. The ANS-5.1-1979 decay heat standard methodology is also a function of average core life, which will decrease for TPC operation since a higher number of fuel assemblies will be replaced each outage (80 feed to as many as 96 feed). The result of decreased average fuel life is a slightly lower normalized decay heat power factor from the tables contained in ANS-5.1-1979. A comparative analysis performed by TVA, utilizing computer code DHEAT, indicated that the increase in reactor core decay energy of a TPC as compared to a non-TPC fuel load would be less than 0.2 MWt. The computer code DHEAT, while based on the ANS-5.1-1979 standard, is an enhanced code which includes other input parameters such as fuel burnup, fuel enrichment, and specific burnup history; therefore, represents a real-time analysis considering both operating an non-operating periods. The small increase determined by TVA's evaluation and the heat load from each TPBAR of less than 3 watts during the cooldown phase is insignificant relative to the overall conservative cooldown assumptions utilized in the Westinghouse cooldown methodology, and the increase represents a small fraction of the total RHR heat load at maximum decay heat (~38 MWt). The existing time period of 29 hours to cool the plant from 350°F to 140°F (2 trains of RHR) and 82 hours (single train of RHR) is not affected by the TPC.

RAI Question 24:

SQN is requesting a number of Technical Specifications associated with Spent Fuel Pool Storage requirements (TS 5.6), including restrictions for each storage region, fuel types which can be stored in each region, acceptable spent fuel loading patterns, limiting burnup requirements by region and fuel type, and other changes. SQN has not submitted any technical justification for these proposed changes. Please provide the technical justification for all of the Spent Fuel Pool Technical Specifications changes being requested. For the proposed changes, include:

a. A summary of applicable design features, licensing basis and relevant regulatory standards and acceptance criteria.

b. A discussion on the analyses performed including a reference to NRC-approved methodology and the applicability of the methodology.

c. Results of the analyses supporting the proposed TS changes and demonstrating that any acceptance criteria and regulatory requirements are satisfied.

Response:

The technical justification for the proposed changes to the spent fuel pool storage requirements is documented in Holtec International Report No. HI-2012629, Revision 01, "Evaluation of the Effect of the TPC on Spent Fuel Storage Criticality Calculations for Sequoyah Nuclear Plant." A copy of this report is in Attachment 3 for NRC review. The methodology used in the report is the same as that used to establish the current spent fuel pool storage requirements approved in response to SQN TS Change Request No. TVA-SQN-TS-99-17. NRC review and approval of the methodology used to establish the current spent fuel pool storage requirements is documented in SERs submitted to TVA by letters dated November 3, 2000 and December 19, 2000.

The report contains information which is considered proprietary by Holtec International. To support receipt of this information by NRC, we have also attached a non-proprietary version of the report in Attachment 4 and enclosed a proprietary data withholding affidavit from Holtec International in Enclosure 3. The affidavit establishes the basis for withholding the proprietary data from public disclosure in accordance with the requirements of 10CFR2.790.

RAI Question 25:

The submittal states that the calculated fluence values were calculated using methods recommended in Regulatory Guide (RG) 1.190. In addition it states that the best estimate values used were determined using a bias factor calculated by comparing calculated surveillance capsule dosimetry. Please clarify:

a. *Whether a staff approved methodology was used for the estimation of the 48 effective full-power years fluence values,*

b. *If the measured dosimetry data used for the estimation of the bias factor were plant specific data, and*

c. *If the peak vessel fluence values calculated for the recent 1.3 percent power uprate were affected by the introduction of the TPBARs.*

Response:

a. The estimation of the 48 effective full power year (EFPY) fluence values was calculated in accordance with RG 1.190. The values were calculated using a synthesis of R-theta, R-Z, and R calculations in accordance with the synthesis equation in the RG. In accordance with RG 1.190, the calculated fluence values are recommended to be used for projection of vessel material properties.

b. Best estimate fluence values were also determined, but these values are recommended to be used only for comparison purposes. The best estimate values are based on the calculated values with a plant-specific bias factor applied. In the case of the SQN units, eight surveillance capsules have been analyzed and the average ratio of measurement to calculation for these capsules is 1.076 for fluence (E > 1.0 MeV). This value falls well within the ±20% tolerance specified in RG 1.190.

c. Introduction of the TPBAR fuel cycles results in a reduction in the neutron leakage at angles symmetrical with the 45° azimuth where the peak vessel fluence occurs. For SQN Unit 1, the reduction in 48 EFPY maximum vessel fluence is calculated to be about 14% and for SQN Unit 2, the reduction is calculated to be about 17%.

RAI Question 26:

Table 3.3-1 in the TPCTR listing 12 functional requirements, how does the licensee address the compliance to these requirements for the TPBAR in 550-effective-full-power day exposure?

TPC TPBAR Functional Requirement 1 from Table 3.3-1: *Structural integrity of the TPC TPBAR shall be maintained throughout Conditions I and II and during shipping and handling.*

Response: TPBAR materials and weldments are selected with adequate strength, creep resistance, fatigue, and fretting wear properties to resist buckling, failure, and damage to associated guide thimbles during a 550 effective full power day (EFPD) exposure, as well as to maintain integrity during shipping and handling. National codes and standards employed by the nuclear industry were used in the TPBAR design and analyses that were conducted to evaluate the performance of the TPBAR during a 550 EFPD exposure.

TPC TPBAR Functional Requirement 2 from Table 3.3-1: *Impact of TPC TPBAR rupture during accident conditions shall be bounded by existing safety analyses limits and offsite / onsite dose shall not exceed 10CFR100 Limits.*

Response: The impact of TPBAR rupture during accident conditions is not affected by an increase from a 520 to a 550 EFPD exposure. The limit on tritium production is not increased by the increased exposure. The contributions from tritium permeation and the release from TPBAR failures is bounded by existing safety analyses limits and offsite/onsite doses do not exceed 10CFR100 limits.

TPC TPBAR Functional Requirement 3 from Table 3.3-1: *Swelling or shrinking of internal TPC TPBAR components shall be accommodated by the design to ensure removability of the TPBARs from the fuel assembly.*

Response: Worst-case tolerances on dimensions were used in the mechanical analysis of the thermal and irradiation expansion to ensure removability of TPBARs from the host fuel assembly after a 550 EFPD exposure. Swelling and shrinking of internal TPBAR components increase as the exposure is increased from 520 to 550

EFPD; however, analyses of deformations due to swelling, irradiation creep, and tensile stresses showed that dimensional changes at 550 EFPD were too small to result in interferences that would compromise the removability of TPBARs from the fuel assembly.

TPC TPBAR Functional Requirement 4 from Table 3.3-1: *The TPC TPBAR cladding stresses and the end plug weld stresses shall not result in cladding collapse, excess ovality, or cracking over the irradiation life of the TPBAR.*

Response: Mechanical analyses of the cladding, end plugs, and weld joint verified margin to failure from collapse, excess ovality, or cracking throughout a 550 EFPD exposure.

TPC TPBAR Functional Requirement 5 from Table 3.3-1: *The cladding shall be free standing and shall not collapse due to external pressure or creep for a design life of 520 EFPD.*

Response: TPBARs are most susceptible to collapse during the initial irradiation period because the internal pressure to resist collapse is lowest at this time of the exposure. Evaluation of thermal and irradiation creep showed that the cladding would not collapse under the coolant pressure and increasing exposures from 520 to 550 EFPD.

TPC TPBAR Functional Requirement 6 from Table 3.3-1: *The TPC TPBAR shall not fail due to vibration fatigue, design cycle fatigue or fretting wear resulting from reactor coolant flow-induced vibration. The host guide thimble shall not fail by fretting wear resulting from reactor coolant flow-induced vibration. The presence of the TPBAR shall not adversely impact the vibration fatigue or design cycle fatigue performance of the host guide thimble.*

Response: TPBAR materials and weldments are selected and designed to resist failure due to vibration fatigue, design cycle fatigue, or fretting wear resulting from reactor coolant flow-induced vibration during an exposure of 550 EFPD. The LTA TPBARs were irradiated to 471 EFPD with no indications of interaction with the associated guide thimbles. Guide thimbles have generally been resistant to fatigue and fretting wear during irradiation. The presence of the TPBAR does not adversely impact the vibration fatigue or design cycle fatigue performance of the host guide thimble. Increasing the EFPD does not change the vibration or the flow characteristics.

TPC TPBAR Functional Requirement 7 from Table 3.3-1: *Corrosion and erosion of the TPC TPBAR outer surface shall not cause material transfer into the reactor coolant in excess of rates comparable with other reactor internal components.*

Response: The corrosion and erosion rate of the outer surface of the TPC TPBAR cladding are negligibly small. Material transfer into the reactor coolant caused by extending the exposure to 550 EFPD is not in excess of rates for other reactor internal components. Corrosion of the LTA TPBAR cladding was negligible after irradiation in WBN to 471 EFPD.

TPC TPBAR Functional Requirement 8 from Table 3.3-1: *The absorber pellet structural integrity shall be maintained over the irradiation life of the TPC TPBAR.*

Response: A conservative pellet gas volume ratio (GVR) limit derived from irradiation test results is applied to ensure the structural integrity of the absorber pellets. The GVR is not dependent on EFPD and is not exceeded during irradiation to 550 EFPD. Therefore, increasing the exposure to 550 EFPD does not represent a change in the evaluation of the structural integrity of the absorber pellets.

TPC TPBAR Functional Requirement 9 from Table 3.3-1: *The plenum spring shall have sufficient preload and spring rate to prevent movement of the pencil column stack during fabrication, shipping, and handling, considering a 4 g axial acceleration loading at beginning of reactor core life.*

Response: The plenum spring (or spring clip) prevents movement of the pencil column stack during fabrication, shipping, and handling. The function of the plenum spring terminates upon completion of the TPBAR insertion into the reactor core. The plenum spring does not have a performance function after the TPBARs are installed into the reactor core. Therefore, increasing the EFPD from 520 to 550 does not affect the function of the plenum spring.

TPC TPBAR Functional Requirement 10 from Table 3.3-1: *The TPC TPBAR shall be sufficiently straight to allow insertion into a fuel assembly and shall maintain dimensional integrity to allow removal from an irradiated fuel assembly without excessive force.*

Response: The primary mechanism that impacts straightness during irradiation is relaxation of localized residual stresses. Most of the residual stresses that affect straightness are relaxed prior to achieving 520 EFPD. Therefore, increasing the EFPD from 520 and 550 does not increase the loss of straightness of TPBARs or compromise the ability to remove irradiated TPBARs from the fuel assemblies without excessive force.

TPC TPBAR Functional Requirement 11 from Table 3.3-1: *The TPC TPBAR shall be similar in its nuclear characteristics to a BPRA and compatible with the nuclear design requirements.*

Response: TPBARs are designed and manufactured to be similar in nuclear characteristics to a BPRA. Increasing the EFPD from 520 to 550 does not change the nuclear characteristics relative to a BPRA.

TPC TPBAR Functional Requirement 12 from Table 3.3-1: *The maximum coolant temperature in a guide thimble containing a TPC TPBAR shall not exceed the coolant bulk boiling temperature during Condition I. (Additional design criteria are specified in Table 3.6-1.)*

Response: Thermal-hydraulic limits do not depend on exposure limits or cycle length. The TPBAR heat generation rate decreases with burnup. Therefore, TPBAR temperatures will decrease during the period of irradiation from 520 to 550 EFPD.

RAI Question 27:

Please address plant specific evaluations required for the TPBARs in a tritium production core as described in Table 3.3-6 of the TPCTR.

Response:

TPC TPBAR Plant Specific Evaluation 1 from Table 3.3-6: *Functional Requirements, verify compliance*

Response: Compliance to functional requirements based on plant specific criteria for WBN and SQN is verified using published data, test results, and analyses. These analyses, which incorporate plant specific input data, have shown adequate margin for meeting the functional requirements. The mapping of the functional/design requirements to the supporting documents is provided in Reference 1. The design requirements matrix indicates how each of the design requirements (i.e., functional requirements) is met in order to ensure that all TPBAR design requirements have been satisfied.

TPC TPBAR Plant Specific Evaluation 2 from Table 3.3-6: *Design Conditions, verify compliance with requirements for:*

- *Production*
- *Cycle Length*
- *Power Density*
- *Power Peaking*
- *Thimble Flow*
- *Check Against Generic Reactor Conditions*

Response: A goal of each TPC design is to maximize the amount of tritium production within the design capabilities of the TPBAR while assuring that all safety criteria are met. Power distribution control in a SQN TPC is accomplished by using a combination of lithium in the fixed TPBAR burnable absorber, and gadolinia as the integral burnable absorber in select fuel rods. For each cycle evaluated in BAW-10237, Revision 1 (SQN TPBAR Topical Report), checks were performed to assure that the minimum and maximum tritium production, including uncertainties, of all TPBARs were within the limits set forth by PNNL. This tritium production check is described in Section 2.4.3 of BAW-10237, Revision 1 (pages 2-11 and 2-12).

TPBARS have a mechanical lifetime of 550 EFPD of exposure. The fuel cycle designs evaluated in the SQNTR were based upon a nominal 510 EFPD length. Designs implemented at SQN will be less than or equal to the 550 EFPD TPBAR design basis.

For each SQN reload core design using TPBARs, the fuel cycle will be designed to be in compliance with the applicable safety criteria while endeavoring to meet the tritium production goals and plant energy requirements. The reload safety evaluation for TPCs will verify that the production limits noted above and the mechanical lifetime of 550 EFPD are satisfied. SQN TPCs will be designed for operation at steady-state RTP with power peaking factors that accommodate operation at the core limits specified in the COLR with adequate margin to the power peaking limits.

The SQN-specific design conditions for core power and core flow rate were used to analysis conditions in the guide thimble and margin to bulk boiling as described in Section 3.6 of BAW-10237, Revision 1. The design conditions used in the analysis are consistent with the current licensing basis that is validated for each reload cycle.

TPC TPBAR Plant Specific Evaluation 3 from Table 3.3-6: *Drawings and Specifications, verify compatibility with assembly design*

Response: TVA has specified the technical, functional, and quality requirements associated with TPBAR irradiation in a TVA nuclear reactor. Included is the requirement of compatibility with the host reactor's fuel assembly design. Framatome-ANP has reviewed the TVA requirements to ensure that its drawings and specifications are in compliance with those requirements. The basis for ensuring compliance with the TVA requirements is that Framatome-ANP's drawings and specifications comply with the production TPBAR design and interface inputs for SQN Units 1 and 2 established by PNNL in PNNL document TTQP-1-118, Revision 6 (Reference 4).

Supporting analyses, utilizing the applicable drawings and specifications, are the same as those identified for functional compliance.

<u>TPC TPBAR Plant Specific Evaluation 4 from Table 3.3-6:</u> *Nuclear Design, verify compliance with and conservatism of input with limits:*

- *Production/Power*
- *Power peaking*

Response: A plant-specific evaluation for SQN was performed for the 96-feed TPBAR cycles evaluated in the SQN TPBAR Topical Report (BAW-10237, Revision 1). Section 2.4.3 presents the results of that evaluation. The predicted tritium production, including uncertainties, was checked against the TPBAR design limits set by PNNL. Limiting core power distributions were modeled using NRC-approved codes and methods (References 2 and 3), and margins to power peaking limits were verified to be acceptable with respect to the core safety and initial condition peaking limits.

The evaluation is described in more detail in Section 2.4.3 of the SQN TPBAR Topical Report, which concluded that SQN reload cores could operate at a thermal power level of 3455 MWt without violating any of the nuclear design basis.

For reload safety evaluations of SQN TPCs, a cycle-specific power distribution analysis will be prepared to demonstrate compliance of the reload core design with respect to power distribution limits. The design will be modeled with nominal lithium concentrations for TPBARs. Based upon the manufacturing tolerances and, as available, the as-built lithium concentrations, power peaking sensitivity to the as-built lithium loading will be determined in a conservative fashion and applied to the power peaking evaluation. The analysis will determine dependence of the core power distribution on fuel exposure, thermal power level, regulating rod position, and transient xenon distribution. The analysis will verify that acceptable peaking margins to the core safety limits are maintained at the reactor trip system setpoints and that acceptable peaking margins to the core initial condition peaking limits are maintained at the normal operating limits. Allowances for calculational uncertainty, engineering hot channel factors, TPBAR helium redistribution due to shutdowns, and the increased peaking due to gaps between TPBAR pencils will be accounted for in the analyses.

Reload safety evaluations also include verification of compliance with SQN UFSAR accident analysis inputs and TSs. With the use of TPBARs, analyses that have power peaking-dependent limits consider calculational uncertainty, engineering hot channel factors, TPBAR helium buildup and the increased power peaking due to gaps between TPBAR pencils. The reactivity impact of the helium buildup and redistribution effect from extended shutdowns will be considered, which may include explicit modeling, or the application of reactivity penalties, if appropriate.

TPC TPBAR Plant Specific Evaluation 5 from Table 3.3-6: *Thermal Hydraulic Evaluation, verify conservatism of conditions for:*

- *Thimble flow and pressure drop*
- *Margin to boiling*

Response: The primary inputs for evaluating the acceptability of the TPBARs residing in the SQN core are the fuel assembly geometry, core flow rate, core power level, heat generation within the guide thimble, and heat generation within the TPBAR. For SQN, a conservative minimum flow geometry was used. The heat generation within the guide tube and TPBAR, conservatively boosted by 15%, are used with design radial peaking adjacent to the thimble tube that reflects a core power of 3455 MWt. In addition, the minimum thermal design flow rate was used that is protected by plant-specific measurement requirements in the TSs. As a result the local coolant temperature predictions and predicted margin to boiling within the thimble tube, determined using the LYNXT code as discussed in Sections 2.4.4 and 3.6 of the SQN TPBAR Topical Report (BAW-10237, Revision 1), reflect a conservative analysis.

TPC TPBAR Plant Specific Evaluation 6 from Table 3.3-6: *Mechanical Performance, verify compliance and conservatism of conditions for:*

- *Tritium and Helium Production, pressure and cladding stress*
- *Pellet GVR limit*
- *Getter loading*
- *Tritium Release*

Tritium and Helium Production, pressure and cladding stress
Response: For each cycle evaluated in the SQN TPBAR Topical Report, checks were performed to assure that the minimum and maximum tritium production, including uncertainties, of all TPBARs were within the limits set forth by Reference 4 to assure the integrity of the TPBAR. This check is described in Section 2.4.3 of BAW-10237, Revision 1 (pages 2-11 and 2-12). The NEMO model includes the ability to predict the helium-3 production and distribution.

Limits on the production of tritium will be checked on a cycle-by-cycle basis. There is a maximum and minimum limit of 1.20 and 0.15 gm. of tritium per rod, respectively. The maximum limit maintains the clad integrity to contain the internal pressure. The minimum limit precludes creep collapse of the TPBAR pin. Because the production of tritium involves the reaction $n+Li^6 \rightarrow$á$+T$ that produces two atoms of gas per atom of Li^6 consumed and T decays to He^3, the internal pressure is primarily driven by the tritium production.

Pellet GVR limit Response: The pellet GVR limit is a single-valued (constant) design limit based on test data for irradiated absorber pellets. This limit does not change based on plant specific evaluations. The pellet GVR was shown to be within the limit for the WBN and SQN operations.

Getter loading Response: The getter loading design limit is a single valued (constant) based on published phase diagrams and test data for irradiated and unirradiated getters. This limit does not change based on plant specific evaluations. The getter loading was shown to be within the limit for the WBN and SQN operations.

Tritium Release Response: TPBAR testing and plant specific analysis have confirmed the design basis of tritium permeation (release) is less than 1000 Ci/1000 TPBARs per year.

References:

1. PNNL-TTQP-1-855 "Production Functional/Design Requirements Mapping to Supporting Documents (U), Pacific Northwest National Laboratory, Richland, Washington, 2001.

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

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

4. TTQP-1-118, Revision 6, "Production TPBAR Design Inputs for Sequoyah Units and 2," Pacific Northwest National Laboratory, Richland, Washington, April 2001.

RAI Question 28:

The consolidation of TPBARs, including related accidents and their potential consequences, were not addressed in NUREG-1672. In Enclosure 4 to their letter of September 21, 2001, the Tennessee Valley Authority (TVA) stated that no more than 24 TPBARs would be damaged for all credible impact scenarios involving a fully-loaded (300 TPBARs) consolidation canister. Based on design features and operating practices that would be applied to handling of consolidation canisters, TVA stated that the maximum credible kinetic energy of a consolidation cannister would be less than that of a dropped fuel assembly and that damage to more than 24 TPBARs was precluded for all credible impact scenarios. Accordingly, the consequences from a fuel handling accident involving a fuel assembly containing an *inventory of 24 TPBARs would bound fuel handling accidents involving a consolidation cannister.*

This approach appears to be neither consistent with regulatory guidance for review of fuel handling facilities (RG 1.13, "Spent Fuel Storage Facility Design Basis;" Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors;" and Sections 9.1.4, 9.4.2, and 15.4.7 of NUREG-0800, "USNRC Standard Review Plan") nor regulatory guidance for review of heavy-load handling systems (NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"). The regulatory guidance for review of fuel handling facilities specifies that the maximum potential release due to an unrestrained drop of a light load from its maximum potential height be evaluated and the resultant consequences are within regulatory limits. The regulatory guidance for review of heavy-load handling systems specifies a complete set of design features and operational controls to ensure reliable performance of the load handling system in preventing damage to important structures, systems, and components. The information in Enclosure 4 to the letter dated September 21, 2001, does not address the maximum potential release from a consolidation cannister, nor does it describe implementation of a complete set of design features and operational controls to ensure reliable performance of the load handling system in preventing damage to important structures, systems, and components.

In order to complete our review, the NRC staff requests that TVA provide either of the following evaluations:

a An evaluation of the maximum potential radiological consequences from a fuel-handling accident involving a consolidation cannister. This evaluation should consider potential releases resulting from an unrestrained drop of a light load from its maximum potential height and address all potential impact combinations involving fuel assemblies and loaded consolidation cannisters.

b An evaluation comparing design features, operational controls, and analyses planned for implementation with those specified in the applicable section of NUREG-0612. This evaluation should address each specified item separately by describing what is planned for implementation and the basis for any difference in scope or depth relative to what is specified in NUREG-0612.

Response:

TVA has chosen to respond to Item "b" above and provides the following evaluation:

NUREG-0612 - Control of Heavy Loads at Nuclear Power Plants

NUREG-0612 provides guidelines to assure that a heavy load drop (heavy load is defined as a load that weighs more than a single spent fuel assembly and its associated handling tool) would not result in a release of radioactive material that could result in off-site doses exceeding 10 CFR Part 100 limits. A heavy load at SQN is 2,100 pounds (lbs). Lifting the TPBAR canister loaded with up to 300 TPBARs is not a heavy load (calculated at approximately 750 lbs buoyant weight), therefore it is not specifically addressed by NUREG-0612. However, in order to provide added assurance that the crane and lifting device used to lift the TPBAR canisters are safe, they will be evaluated against the requirements of NUREG-0612.

The Spent Fuel Bridge Crane will be the only crane utilized to lift the TPBAR canister while loaded with TPBARs. The bridge itself is designed specifically by Dwight Foote, Inc. for the provided hoist (2000 lb capacity hoist). Any reference to the crane or crane attributes such as trolley, bridge, hoist, etc. pertain specifically to the Spent Fuel Bridge Crane unless otherwise indicated.

In Section 5.1.1 of NUREG-0612, general requirements are outlined for handling of heavy loads. The applicable portions of these requirements and TVA's response are as follows:

1. **Safe load paths - a defined path should be established for movement of heavy loads that minimizes the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel or spent fuel pool, or to impact safe shutdown equipment.**

   Response: TPBAR canister has been evaluated for uncontrolled (40 feet per minute [fpm] maximum) lowering and no damage to TPBARs will occur as demonstrated by impact analysis as previously discussed in TVA letter dated February 21, 2002. The loaded canister weighs less than a fuel assembly and therefore damage to stored spent fuel from an uncontrolled canister lowering is bounded by existing analysis of a fuel handling accident.

   Additionally, loaded canister movement is restricted to the area within the Spent Fuel Pool and Cask Loading Pit which precludes interaction with safety-related equipment. Loaded canisters will be stored in designated cells in the SFP away from anticipated fuel assembly movement. Additional administrative controls will be in place to prevent handling fuel assemblies over these cells while loaded canisters are present. Therefore, all load paths for this crane are considered safe and do not require designation.

2. **<u>Procedures</u> - should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment.**

   Response: Appropriate detailed procedures will be developed to address load handling operations of the Spent Fuel Bridge Hoist to lift the TPBAR canister.

3. **<u>Crane Operators</u> - should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, "Overhead and Gantry Cranes."**

   Response: Crane Operators are trained, qualified, and conduct themselves in accordance with ASME B30.2. Additionally, TPBAR consolidation operators will be required to have the same training needed to perform fuel handling activities.

4. **<u>Special Lifting Devices</u> - should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 lb. (4500 Kg) or More for Nuclear Materials."**

   Response: The lifting device for the TPBAR canister will be designed to satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 lb. (4500 Kg) or More for Nuclear Materials." Specifically, either dual load paths or increased safety factors, in addition to fabrication and testing requirements, will be invoked in accordance with Sections 6 and 7 of ANSI N14.6.

5. **<u>Lifting Devices that are not specially designed</u> - should be installed and used in accordance with the guidelines of ANSI B30.9-1971 "Slings."**

   Response: No slings will be utilized to lift the TPBAR canister. However, a synthetic sling is utilized as a lanyard to limit canister tipping to prevent TPBAR spillage. This lanyard is designed to withstand the impulse/impact load to stop the tipping canister.

6. **The <u>crane</u> - should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, "Overhead and Gantry Cranes."**

   Response: The SFP crane will be inspected, tested, and maintained prior to each refueling outage. The TPBAR consolidation activity will be performed, when necessary, following plant startup after each refueling outage. The SFP crane maintenance procedure prescribes inspection and maintenance required for this crane. Further, other site procedures govern operator conduct and load handling per ASME B30.2.

7. **The crane should be designed to meet the applicable criteria and guidelines of Chapter 1 of ANSI B30.2-1976, "Overhead and Gantry Cranes" of Crane Manufacturers Association of America (CMAA)-70, "Specifications for Electric Overhead Traveling Cranes." An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied.**

   Response: The hoist was designed in accordance with ANSI B30.16 (Applicable Standard at the time of hoist design and fabrication). Note that the importance of the structural elements contained in the required specifications is diminished as the maximum critical load (MCL) is less than one half of the crane's capacity.

   Additionally, the crane used for handling the TPBAR canister will be compared to single-failure proof guidelines to assure increased safety while performing this lift. Single Failure Proof Requirements for Nuclear Power Plant Cranes are contained in NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants."

NUREG-0554 - Single-Failure-Proof Cranes for Nuclear Power Plants

Single failure proof guidelines are outlined in NUREG-0554. A comparison of the hoist and bridge crane and single failure proof requirements from the applicable section of NUREG-0554 is provided below. The applicable portions of NUREG-0554 used to document the requirement is also included:

2.0 Specification and Design Criteria

2.1 Construction and Operating Periods

Requirement: Separate performance specifications for a crane system may be needed to reflect the duty cycles and loading requirements for construction phase and operating plant phase.

Response: The SFP bridge hoist was not used extensively during construction. The limited range of the crane (could only perform lifts within the spent fuel pit) and the availability of the 125/10 ton refuel floor bridge crane, which could cover the entire refuel floor, made it impractical for construction use. Therefore, construction phase duty and loading cycles are not a concern. The duty cycles and loading requirements for the operating phase are defined.

2.2 Maximum Critical Load

Requirement: A single failure proof crane should be designed to handle the MCL that will be imposed. Certain single failure proof cranes may be required to handle occasional non-critical loads greater than the MCL. The maximum non-critical load will be the design rated load (DRL). The DRL and the MCL ratings should be marked on the crane separately.

Response: The MCL that will be imposed consists of the TPBAR canister and up to 300 TPBARs, which will weigh approximately 750 lbs. This is well within the capacity of the hoist including any dynamic loading (DRL = 1 ton). Since the consolidation canister is the only critical load and the DRL is the hoist capacity which is marked on the hoist, no additional markings are deemed necessary.

2.3 Operating Environment

Requirement: The operating environment, including maximum and minimum pressure, maximum rate of pressure increase, temperature, humidity, and emergency corrosive or hazardous conditions, should be specified for the crane and lifting fixtures.

Response: The normal range of minimum and maximum temperatures on the refuel floor is 60°F to 104°F. Pressure is maintained slightly below atmospheric. Relative humidity is maintained between 30% and 90%. There are no emergency corrosive or hazardous conditions. Further, lifting devices are designed to withstand the aqueous conditions within the SFP.

2.4 Material Properties

Requirement: Cranes are generally fabricated from structural shapes and plate rolled from carbon steel (no alloying elements except for 1% manganese in heavier section) or low alloy steel (less than 5% total alloy content). Some of these steel parts exceed 12 millimeters (1/2 inch) in thickness and may have brittle-fracture tendencies when exposed to low operating temperatures so that testing of the material toughness becomes necessary. When low-alloy steels are used, weld metal toughness is of greater concern than the base metal. The crane and lifting fixtures for cranes already fabricated or operating may be subjected to a cold proof test.

Response: This requirement is written concerning brittle fracture tendencies of structural steel that exceeds 1/2 inch thickness when exposed to lower operating temperatures. The crane is located indoors, in a controlled environment, and not subject to extremes in temperature. Therefore, is not considered necessary to perform a fracture analysis to determine the minimum operating temperature.

2.5 Seismic Design

Requirement: The cranes should be designed to retain control of and hold the load, and the bridge and trolley should be designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event. If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, the bridge should remain on the runway with brakes applied, and the trolley should remain on the crane girders with brakes applied.

Response: The bridge and hoist have been evaluated for seismic loading (with a fuel assembly which is heavier) and are acceptable.

2.6 Lamellar Tearing

Requirement: All weld joints whose failure could result in the drop of a critical load should be nondestructively examined. If any of these weld joint geometrics would be susceptible to Lamellar tearing, the base metal at the joints should be nondestructively examined.

Response: This hoist is rated for loads in excess of 2 1/2 times (Factor of Safety of approximately 13) the MCL, and has not experienced problems with lifting heavier loads. An inspection is performed periodically on the crane (prior to refueling outages) to check for cracks or distortion; therefore, Lamellar tearing will not be a problem while lifting an MCL.

2.7 Structural Fatigue

Requirement: Since each crane loading cycle will produce cyclic stress, it may be necessary to investigate the potential for failure of the metal due to fatigue. If a crane will be used during the construction period, it will experience additional cyclic loading, and these loads should be added to the expected cyclic loading for the permanent plant operation when performing the fatigue evaluation.

Response: The SFP crane was used sparingly during construction because of its limited range (only can be used to make lifts in the spent fuel pool) as compared to the 125/10 ton overhead bridge crane, which can access almost all of the refuel floor. This crane is currently used to move fuel only a few times per year (usually for refueling outages), and has not and will not receive the volume of cyclic loading that might require a structural fatigue analysis.

2.8 Welding Procedures

Requirement: Preheat temperatures and postweld heat-treatment (stress relief) temperatures for all weldments should be specified in the weld procedure. Welds described in the recommendations of Section 2.6 should be postweld heat treated in accordance with Subarticle 3.9 of AWS D1.1, "Structural Welding Code."

Response: The SFP crane has been in use for several years with no identified welding problems, and is visually inspected periodically for problems with welds. Therefore it is acceptable to use for MCL lifts of less than half the crane's capacity.

3.0 Safety Features

3.1 General

No response is applicable.

3.2 Auxiliary Systems

Requirement: All auxiliary hoisting systems of the main crane handling system that are employed to lift or assist in handling critical loads should be single failure proof.

Auxiliary systems or dual components should be provided for the main hoisting mechanism so that, in case of subsystem or component failure, the load will be retained and held in a stable or immobile safe position.

Response: The hoist on this crane has dual braking. If there is a loss of power, a mechanical brake will hold the load in place. The factors of safety for this hoist is in excess of 13 to 1. Therefore the SFP crane has a high factor of safety while lifting the MCL which assures safe handling of critical loads, and a dual braking system, which assures that the load will be retained in a stable and immobile safe position in case of a component failure.

3.3 Electric Control Systems

Requirement: The automatic controls and limiting devices should be designed so that, when disorders due to inadvertent operator action, component malfunction, or disarrangement of subsystem control functions occur singly or in combination during the load handling, and assuming no components have failed in any subsystems, these disorders will not prevent the handling system from stopping and holding the load. An emergency stop button should be added at the control station to stop all motion.

Response: There are redundant upper limit switches of different designs to stop the hoisting in the up direction and prevent two-blocking. Simultaneous hoist and bridge operation is precluded by interlocks. The trolley is manual. Therefore, uncontrolled lowering is considered the only plausible control failure consequence. Uncontrolled lowering of the TPBAR canister has been evaluated and demonstrates that no TPBAR damage occurs at a hoist speed of 40 fpm (currently maximum hoist speed is 20 fpm) for all potential impact scenarios. A lanyard is installed during hoisting to assure TPBARs are not spilled out of the canister in the event of canister tipping following impacting an obstruction. Further, the canister and its handling tool weighs less than a spent fuel assembly and its handling tool, therefore, consequences of this is bounded by existing FHA analysis.

3.4 Emergency Repairs

Requirement: A crane that has been immobilized because of malfunction or failure of controls or components while holding a critical load should be able to hold the load or set the load down while repairs or adjustments are made. This can be accomplished by inclusion of features that will permit manual operation of the hoisting system and the bridge and trolley transfer mechanisms by means of appropriate emergency devices.

Means should be provided for using the devices required in repairing, adjusting, or replacing the failed components(s) or subsystem(s) when failure of an active component or subsystem has occurred and the load is supported and retained in the safe (temporary) position with the handling system immobile. As an alternative to repairing the crane in place, means may be provided for safely transferring the immobilized system with its load to a safe laydown area that has been designated to accept the load while the repairs are being made.

The design of the crane and its operating area should include provisions that will not impair the safe operation or safe shutdown of the reactor or cause unacceptable release of radioactivity when corrective repairs, replacements, and adjustments are being made to place the crane handling system back into service after component failure(s).

Response: Access to the Spent Fuel Bridge in order to repair the hoist and the ability to take measures to assure the load will be retained in a safe, temporary position will not be a concern because the Spent Fuel Bridge is located on the refuel floor, with easy personnel access at any location in its travel. It would be relatively easy to take measures to retain the TPBAR canister in place (by using a sling or another hoist/crane such as the Auxiliary Building Bridge Crane) with a minimum factor of safety of 10-1) because of its accessibility to personnel and because the load is relatively light (750 lbs). The TPBAR canister must be in the spent fuel pool as long as it contains TPBARs; therefore a safe laydown area would be limited to the spent fuel racks.

If the hoist/load becomes immobilized due to a hoist malfunction, the load could be temporarily rigged and either suspended in place or placed in a spent fuel rack utilizing another hoist (with a factor of safety of 10 to 1 minimum) while the original hoist is being repaired. If the trolley or bridge travel is affected, the hoist will be able to retain the load while repairs are in progress.

4.1 Reeving System

Requirement: Component parts of the vertical hoisting mechanism are important. Specifically, the rope reeving system deserves special consideration during design of the system. The load-carrying rope will suffer accelerated wear if it rubs exclusively on the sides of the grooves in the drum and sheaves because of improper alignment or large fleet angles between the grooves. The load-carrying rope will furthermore suffer excessive loading if it is partly held by friction on the groove wall and then suddenly released to enter the bottom of the groove. The rope can be protected by the selection of conservative fleet angles. Ropes may also suffer damage due to excessive strain developed if the rope construction and the pitch diameter of the sheaves are not properly selected. Fatigue stress in ropes can be minimized when the pitch diameter of the sheaves is selected large enough to produce only nominal stress levels. The pitch diameter of the sheaves should be larger for ropes moving at the highest velocity near the drum and can be smaller for sheaves used as equalizers where the rope is stationary. Protection against excessive wire rope wear and fatigue damage can be ensured through scheduled inspection and maintenance.

Design of the rope reeving system(s) should be dual with each system providing separately the load balance on the head and load blocks through configuration of ropes and rope equalizer(s). Selection of the hoisting rope or running rope should include consideration of the size, construction, lay, and means or type of lubrication, if required, to maintain efficient working of the individual wire strands when each section of rope passes over the individual sheaves during the hoisting operation. The effects of impact loadings, acceleration, and emergency stops should be included in selection of rope reeving systems. The maximum load (including static and inertia forces) on each individual wire rope in the dual reeving system with the MCL attached should not exceed 10% of the manufacturer's published breaking strength.

The ratio of wire rope yield strength to ultimate strength may vary sufficiently for different production runs to influence the wire rope rating in such a manner that the initial safety margin selected would be too small to prevent the critical load from straining the wire rope material beyond the yield point under abnormal conditions. It would, therefore, be prudent to consider the wire rope yield strength as well as the ultimate strength when specifying wire rope in order to ensure the desired margin on rope strength.

The maximum fleet angle from drum to the lead sheave in the load block or between individual sheaves should not exceed 0.061 Rad (3-1/2°) at any one point during hoisting except that for the last 1 meter (3 feet) of maximum lift elevation the fleet angle may increase slightly. The use of reverse bends for running wire ropes should be limited, and the use of larger sheaves should be considered for those applications where a disproportionate reduction in wire rope fatigue life would be expected from the use of standard sheave diameters for reverse bends.

The equalizer for stretch and load on the rope reeving system may be of either beam or sheave type or combinations thereof. A dual rope reeving system with individual attaching points and means for balancing or distributing the load between the two operating rope reeving systems will permit either rope system to hold the critical load and transfer the critical load without excessive shock in case of failure of the other rope system.

The pitch diameter of running sheaves and drums should be selected in accordance with the recommendations of CMAA Specification #70. The dual reeving system may be a single rope from each end of a drum terminating at one of the blocks or equalizer with provisions for equalizing beam-type load and rope stretch, with each rope designed for the total load. Alternatively, a 2-rope system may be used from each drum or separate drums using a sheave equalizer or beam equalizer or any other combination that provides two separate and complete reeving systems.

Response: The wire rope on this hoist is regularly inspected in accordance with site procedures. Accordingly, excessive wire rope wear and fatigue damage are not a concern. The reeving system on this hoist is not dual; however, the factor of safety while lifting the MCL will be approximately 13 to 1. With this high factor of safety, the reeving will have an acceptable breaking strength.

The hoist for the spent fuel pit bridge crane incorporates a sheave type equalizer to assure that the load in the reeving system will be equally distributed by compensating for rope stretch or swinging of the block.

4.2 Drum Support

Requirement: The load hoisting drum should be provided with structural and mechanical safety devices to limit the drop of the drum and thereby prevent it from disengaging from its holding brake system if the drum shaft or bearings were to fail or fracture.

Response: While the hoist does not meet these requirements, the increased factor of safety (13 to 1) while lifting the MCL, as well as the fuel handling activities which precede consolidation activities, makes it very unlikely that the load hoisting drum will fail.

4.3 Head and Load Blocks

Requirement: The head and load blocks should be designed to maintain a vertical load balance about the center of lift from load block through head block and have a reeving system of dual design.

The load-block assembly should be provided with two load-attaching points (hooks or other means) so designed that each attaching point will be able to support a load of three times the load (static and dynamic) being handled without permanent deformation of any part of the load-block assembly other than local strain concentration in areas for which additional material has been provided for wear.

The individual component parts of the vertical hoisting system components, which include the head block, rope, reeving system, load block, and dual load-attaching device, should each be designed to support a static load of 200% of the MCL. A 200% static type load test should be performed for each load-attaching hook. Measurements of the geometric configuration of the hooks should be made before and after the test and should be followed by a nondestructive examination that should consist of volumetric and surface examinations to verify the soundness of fabrication and ensure the integrity of the hooks. The load blocks should be nondestructively examined by surface and volumetric techniques. The results of the examinations should be documented and recorded.

Response: While the hoist does not have a reeving system of dual design, and the load-block assembly is not provided with two load-attaching points, the factor of safety of this hoisting system for the MCL is in excess of 13 to 1 and is deemed acceptable. The hoist and crane are visually inspected at regular intervals, and the results are documented in accordance with procedure.

4.4 Hoisting Speed

Requirement: Maximum hoisting speed for the critical load should be limited to that given in the "slow" column of Figure 70-6 of CMAA Specification #70.

Selection of hoisting speed is influenced by such items as reaction time for corrective action for the hoisting movement and the potential behavior of a failed rope. To prevent or limit damaging effects that may result from dangerous rope spin-off in case of a rope break, the hoisting speed should be limited. The rope traveling speed at the drum is higher than at other points in the reeving system, and the potential for damage due to rope failing and interference with other parts of the system should be considered. Conservative industry practice limits the rope line speed to 1/4 meter per second (50 fpm) at the drum.

Response: Rope line speed is less than 50 fpm. Additionally, adverse inertial affects are diminished due to the MCL being less than 1/2 of the rated load.

4.5 Design Against Two-Blocking

Requirement: A potential failure of a hoist travel-limit switch could result in a "two-block" incident and in the cutting or crushing of the wire rope. In order to protect the wire rope, the reeving system should be designed to prevent the cutting or crushing of the wire rope if a two-blocking incident were to occur.

The mechanical and structural components of the complete hoisting system should have the required strength to resist failure if the hoisting system should "two-block" or if "load hang-up" should occur during hoisting. The designer should provide means within the reeving system located on the head or on the load-block combinations to absorb or control the kinetic energy of rotating machinery during the incident of two-blocking. As an alternative, the protective control system to prevent the hoisting system from two-blocking should include, as a minimum, two independent travel limit switches of different designs and activated by separate mechanical means. These devices should de-energize the hoist drive motor and the main power supply. The protective control system for load hang-up, a part of the overload protection system, should consist of load cell systems in the drive train or motor-current-sensing devices or mechanical load-limiting devices. The location of mechanical holding brakes and their controls should provide positive, reliable, and capable means to stop and hold the hoisting drum(s) for the condition described in the design specification and in this recommendation. This should include capability to withstand the maximum torque of the driving motor if a malfunction occurs and power to the driving motor cannot be shut off. The auxiliary hoist, if supplied, should be equipped with two independent travel-limit switches to prevent two-blocking.

Response: The SFP Bridge Crane has both a weighted mechanical limit switch and a geared limit switch to stop upward motion of the hoist. The hoist has a load monitor/limiter to assure that the hoist is not subjected to a load hang-up. The limit switches and load monitoring features are verified for proper operation prior to each refueling outage.

4.6 Lifting Devices

Requirement: Lifting devices that are attached to the load block such as lifting beams, yokes, ladle or trunnion-type hooks, slings, toggles, and clevises should be conservatively designed with a dual or auxiliary device or combinations thereof. Each device should be designed or selected to support a load of three times the load (static and dynamic) being handled without permanent deformation.

Response: The special lifting device used to lift the MCL will meet applicable requirements of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 lbs. (4500 Kg) or More for Nuclear Materials."

4.7 Wire Rope Protection

Requirement: Side loads would be generated to the reeving system if hoisting were done at angles departing from a normal vertical lift and resulting damage could be incurred in the form of excessive wear on sheaves and wire rope. A potential would also exist for the wire rope to be cut by jumping its groove barrier on the drum. If side loads cannot be avoided, the reeving system should be equipped with a guard that would keep the wire rope properly located in the grooves on the drum.

Response: This SFP crane is used to lift spent fuel bundles and will be used in the future to lift the TPBAR canisters. The bridge crane is designed to provide control to raise and lower spent fuel into the racks. The design of the handling tool and the required crane alignment necessary to engage the canister precludes side loading. Therefore, no special guard will be required on the hoist reeving.

4.8 Machinery Alignment

Requirement: Power transmission gear trains are often supported by fabricated weldments of structural parts. The proper alignment of shafts and gears depends on the adequacy of bearings and their supports to maintain correct alignment of all components. The proper functioning of the hoisting machinery during load handling can best be ensured by providing adequate support strength of the individual component parts and the welds or bolting that binds them together. Where gear trains are interposed between the holding brakes and the hoisting drum, these gear trains should be single failure proof and should be of dual design.

Response: This hoist was constructed as a production package by an experienced manufacturer. This hoist has been utilized for many years without internal hoist package alignment problems.

Additionally, since the alignment issue is related to structural adequacy and the MCL is less than 1/2 of the hoist capacity, the potential for malfunctions due to misalignment are negligible.

4.9 Hoist Braking System

Requirement: Mechanical holding brakes in the hoisting system (raising and lowering) that are automatically activated when electric power is off or mechanically tripped by overspeed devices or overload devices in the hoisting system will help ensure that a critical load will be safely held or controlled in case of failure in the individual load-bearing parts of the hoisting machinery.

Each holding brake should have more than full-load stopping capacity but should not have excessive capacity that could cause damage through sudden stopping of the hoisting machinery. A minimum brake capacity of 125% of the torque developed during the hoisting operation at the point of brake application has been determined to be acceptable.

The minimum hoisting braking system should included one power control braking system (not mechanical or drag brake type) and two holding brakes. The holding brakes should be applied when power is off and should be automatically applied on overspeed to the full holding position if a malfunction occurs. Each holding brake should have a torque rating not less than 125% of the full-load hoisting torque at point of application (location of the brake in the mechanical drive). The minimum number of braking systems that should be operable for emergency lowering after a single brake failure should be two holding brakes for stopping and controlling drum rotation.

The holding brake system should be single failure proof; i.e., any component or gear train should be dual if interposed between the holding brakes and the hoisting drums. The dynamic and static alignment of all hoisting machinery components, including gearing, shafting, couplings, and bearings, should be maintained throughout the range of loads to be lifted, with all components positioned and anchored on the trolley machinery platform.

Manual operation of the holding brakes may be necessary during an emergency condition, and provision for this should be included in the design conditions. Adequate heat dissipation from the brake should be insured so that damage does not occur if the lowering velocity is permitted to increase excessively. It may be necessary to stop the lowering operation periodically to prevent overheating and permit the brake to dissipate the excess heat.

Portable instruments should be used to indicate the lowering speed during emergency operations. If a malfunction of a holding brake were to occur and emergency lowering of the load become necessary, the holding brake should be restored to working condition before any lowering is started.

Response: The hoist has both a direct acting magnetic brake to stop rotation when the power is off, and a disc type brake to stop the load when desired. Also, since the MCL is less than 1/2 of the hook capacity, the braking system is significantly oversized for this lift.

5.0 Bridge and Trolley

5.1 Braking Capacity

Requirement: Failure of the bridge and trolley travel to stop when power is shut off could result in uncontrolled incidents. This would be prevented if both bridge and trolley drives are provided with control and holding braking systems that would be automatically applied when the power is shut off or if an overspeed or overload condition occurs because of malfunction or failure in the drive system.

To avoid the possibility of drive motor overtorque within the control system, the maximum torque capability of the driving motor and gear reducer for trolley motion and bridge motion of the overhead bridge crane should not exceed the capability of gear train and brakes to stop the trolley or bridge from the maximum speed with the DRL attached. Incremental or fractional inch movements, when required, should be provided by such items as variable speed controls or inching motor drives. Control and holding brakes should each be rated at 100% of maximum drive torque that can be developed at the point of application. If two mechanical brakes, one for control and one for holding, are provided, they should be adjusted with one brake in each system leading the other and should be activated by release or shutoff of power. This applies to both trolley and bridge. The brakes should also be mechanically tripped to the "on" "holding" position in the event of a malfunction in the power supply or an overspeed condition. Provisions should be made for manual emergency operation of the brakes. The holding brake should be designed so that it cannot be used as a foot-operated slowdown brake. Drag brakes should not be used. Mechanical drag-type brakes are subject to excessive wear, and the need for frequent service and repair tends to make this type of brake less reliable; they should therefore not be used to control movements of the bridge and trolley.

Opposite-driven wheels on bridge or trolley that support bridge or trolley on their runways should be matched and should have identical diameters.

Trolley and bridge speed should be limited. The speed limits indicated for slow operating speeds for trolley and bridge in specification CMAA #70 are recommended for handling MCLs.

Response: The trolley operation is a manual chain drive; therefore, there are no loss of power, torque, braking, over-speed, overload or operating speed issues associated with the trolley.

The bridge drive is two speed (11 - 28 fpm). End stops are provided for both the bridge and trolley. Because the trolley is manual, no trolley brakes are required. Bridge and hoist movement is provided with an interlock. The MCL is less than one half of the crane capacity, thereby reducing braking requirements. Additionally, because the TPBARs are protected by the canister, in the highly unlikely event that the bridge drives it into the SFP wall or other structure, braking issues are not of major concern for TPBAR protection.

5.2 Safety Stops

Requirement: Limiting devices, mechanical and/or electrical, should be provided to control or prevent overtravel and overspeed of the trolley and bridge. Buffers for bridge and trolley travel should be included at the ends of the rails.

Safety devices such as limit-type switches provided for malfunction, inadvertent operator action, or failure should be in addition to and separate from the limiting means or control devices provided for operation.

Response: Both Bridge and Trolley have vendor supplied bridge and trolley stops.

6.0 Drivers and Controls

6.1 Driver Selection

Requirement: The horsepower rating of the hoist driving motor should be matched with the calculated requirement that includes the design load and acceleration to the design hoisting speed. Overpowering of the hoisting equipment would impose additional strain on the machinery and load-carrying devices by increasing the hoisting acceleration rate.

To preclude excessive drive motor torque, the maximum torque capability of the electric motor drive for hoisting should not exceed the rating or capability of the individual components of the hoisting system required to hoist the MCL at the maximum hoisting speed. Overpower and overspeed conditions should be considered an operating hazard as they may increase the hazard of malfunction or inadvertent operation. It is essential that the controls be capable of stopping the hoisting movement within amounts of movement that damage would not occur. A maximum hoisting movement of 8 cm (3 inches) would be an acceptable stopping distance.

Normally a crane system is equipped with mechanical and electrical limiting devices to shut off power to driving motors when the crane hook approaches the end of travel or when other parts of the crane system would be damaged if power were not shut off. It is prudent to include safety devices in the control system for the crane, in addition to the limiting devices, for the purpose of ensuring that the controls will return to or maintain a safe holding position in case of malfunction. Electric circuitry design should be carefully considered so that the controls and safety devices ensure safe holding of the critical load when called upon to perform their safety function. For elaborate control systems, radio control, or ultimate control under unforeseen conditions of distress, an "emergency stop button" should be placed at ground level to remove power from the crane independently of the crane controls. For cranes with a DRL rating much higher than the MCL rating, it may be necessary to provide electrical or mechanical resetting of overload sensing devices when changing from one operation to the other. Such resetting should be made away from the operator cab location and should be included in an administrative control program.

Response: The hoist motor was sized to lift spent fuel bundles, which weigh approximately 2000 lbs. The hoist is a standard package supplied by a vendor for the DRL. As a result, drivers are considered oversized for the MCL and are considered acceptable. Resetting of the load sensing device will be required and procedurally controlled when switching between fuel handling and TPBAR consolidation evolutions.

6.2 Driver Control Systems

Requirement: The control systems should be designed as a combination of electrical and mechanical systems and may include such items as contactors, relays, resistors, and thyristors in combination with mechanical devices and mechanical braking systems. The control system(s) provided should include consideration of the hoisting (raising and lowering) of all loads, including the rated load, and the effects of the inertia of the rotating hoisting machinery such as motor armature, shafting and coupling, gear reducer, and drum. If the crane is to be used for lifting spent fuel elements, the control system should be adaptable to include interlocks that will prevent trolley and bridge movements while the load is being hoisted free of a reactor vessel or a storage rack, as may be recommended in RG 1.13, "Spent Fuel Storage Facility Design Basis."

Response: The control system provided with this SFP crane was designed for hoisting loads in the spent fuel pit. The bridge drive and the hoist are interlocked on this crane to prohibit simultaneous operation of the bridge and hoist. The crane system is designed to lift the weight of fuel bundles, and is of sufficient capacity to make these lifts. It is also of sufficient capacity to perform the TPBAR canister lift.

6.3 Malfunction Protection

Requirement: Means should be provided in the motor control circuits to sense and respond to such items as excessive electric current, excessive motor temperature, overspeed, overload, and overtravel. Controls should be provided to absorb the kinetic energy of the rotating machinery and stop the hoisting movement reliably and safely through a combination of electrical power controls and mechanical braking systems and torque controls if one rope or one of the dual reeving systems should fail or if overloading or an overspeed condition should occur.

Response: The SFP crane is a standard hoist package from an experienced vendor. Overload protection, etc., is commensurate with requirements of ANSI B30.16. Furthermore, since the MCL is less than one half of the hook capacity and the crane routinely handles much heavier loads, these protective features are less significant.

6.4 Slow Speed Drives

Requirement: Increment drives for hoisting may be provided by stepless controls or inching motor drive. If jogging or plugging is to be used, the control circuit should include features to prevent abrupt change in motion. Drift point in the electric power system when provided for bridge or trolley movement should be provided only for the lowest operating speeds.

Response: The SFP crane has been designed for fuel handling. As such, it is well suited to handling the lighter TPBAR consolidation canister between the SFP racks, the consolidation fixture, or the transportation cask. Travel speeds, jogging functions, etc., needed for consolidation are compatible with those needed for fuel handling activities.

6.5 Safety Devices

Requirement: Safety devices such as limit-type switches provided for malfunction, inadvertent operator action, or failure should be in addition to and separate from the limiting means or control devices provided for operation.

Response: The additional safety feature of the analyzed protective canister, lifting device with increased safety factors, additional administrative limitations, and the handling lanyard are in addition to the limiting means or control devices provided for normal crane operation.

6.6 Control Stations

Requirement: The complete operating control system and provisions for emergency controls for the overhead crane handling system should preferably be located in a cab on the bridge. When additional operator stations are considered, they should have control systems similar to the main station. Manual controls for hoisting and trolley movement may be provided on the trolley. Manual controls for the bridge may be located on the bridge. Remote control or pendant control for any of these motions should be identical to those provided on the bridge cab control panel. Cranes that use more than one control station should be provided with electrical interlocks that permit only one control station to be operable at any one time. In the design of the control systems, provision for and locations of devices for control during emergency conditions should be provided.

Response: This requirement is for a crane with a cab. Because the crane does not have a cab or multiple control stations, this requirement is not applicable.

7.0 Installation Instructions

7.1 General

Requirement: Installation instructions should be provided by the manufacturer. These should include a full explanation of the crane handling system, its controls, and the limitations for the system and should cover the requirements for installation, testing, and preparations for operation.

Response: The crane has been installed for several years. The vendor submitted technical drawings and Operation Manuals to explain the above.

7.2 Construction and Operating Periods

Requirement: When the permanent plant crane is to be used for construction and the operating requirements for construction are more severe than those required for permanent plant service, the construction operating requirements should be defined separately. The crane should be designed structurally and mechanically for the construction loads, plant service loads, and their functional performance requirements. At the end of the construction period, the crane handling system should be modified as needed for the performance requirements of the nuclear power plant operating service. After construction use, the crane should be thoroughly inspected by nondestructive examination and load tested for the operating phase. The extent of nondestructive examination, the procedures used, and the acceptance criteria should be defined in the design specification. If allowable design stress limits for the plant operating service are to be exceeded during the construction phase, added inspection supplementing that described in Section 2.6 should be specified and developed.

During and after installation of the crane, the proper assembly of electrical and structural components should be verified as to satisfaction of installation and design requirements.

Response: This SFP crane was used sparingly during construction because of its limited range and capacity (only can be used to make lifts in the spent fuel pool). Additionally, any use of this crane during construction was consistent with use during fuel handling operations. As a result, no additional requirements, examinations, or modifications are warranted.

8.0 Testing and Preventive Maintenance

8.1 General

Requirement: A complete check should be made of all the crane's mechanical and electrical systems to verify the proper installation and to prepare the crane for testing.

Information concerning proof testing on components and subsystems that was required and performed at the manufacturer's plant to verify the ability of components or subsystems to perform should be available for the checking and testing performed at the place of installation of the crane system.

Response: The SFP crane/hoist have been in service for years and are operating normally. Proper operation and crane condition is verified prior to each refueling outage.

8.2 Static and Dynamic Load Tests

Requirement: The crane system should be static load tested at 125% of the MCL. The tests should include all positions generating maximum strain in the bridge and trolley structures and other positions as recommended by the designer and manufacturer. After satisfactory completion of the 125% static test and adjustments required as a result of the test, the crane handling system should be given full performance tests with 100% of the MCL for all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices. The features provided for manual lowering of the load and manual movement of the bridge and trolley during an emergency should be tested with the MCL attached to demonstrate the ability to function as intended.

Response: The crane routinely lifts approximately 2000 lbs during refueling outages. It is procedurally checked out prior to outages and inspected. Since the crane is designed for more than double the MCL, and since it is routinely inspected at regular intervals, it is acceptable without further testing.

8.3 Two-Block Test

Requirement: When equipped with an energy-controlling device between the load and head blocks, the complete hoisting machinery should be allowed to two-block during the hoisting test (load-block limit and safety devices are bypassed). This test, conducted at slow speed without load, should provide assurance of the integrity of the design, the equipment, the controls, and the overload protection devices. The test should demonstrate that the maximum torque that can be developed by the driving system, including the inertia of the rotating parts at the overtorque condition, will be absorbed or controlled during a two-blocking or load hang-up. The complete hoisting machinery should be tested for ability to sustain a load hang-up condition by a test in which the load-block attaching points are secured in a fixed anchor or excessive load. The crane manufacturer may suggest additional or substitute test procedures that will ensure the proper functioning of protective overload devices.

Response: The hoist is not equipped with energy controlling devices; therefore, a two-block test would be unacceptable. This hoist utilizes a load monitor/limiter to assure that any load hang-up will not damage the crane. Additionally, the hoist is equipped with dual limit switches to assure that it does not two-block.

8.4 Operational Tests

Requirement: Operational tests of crane systems should be performed to verify the proper functioning of limit switches and other safety devices and the ability to perform as designed. However, special arrangements may have to be made to test overload and overspeed sensing devices.

Response: The SFP crane has been installed and operating adequately for years. Proper functioning and condition of components associated with the crane are verified periodically by procedural testing.

8.5 Maintenance

Requirement: After installation, equipment usually suffers degradation due to use and exposure. A certain degree of wear on such moving parts as wire ropes, gearing, bearings, and brakes will reduce the original design factors and the capacity of the equipment to handle the rated load. With good maintenance practice, degradation is not expected to exceed 15% of the design load rating, and periodic inspection coupled with a maintenance program should ensure that the crane is restored to the design condition if such degradation is found. Essentially, the MCL rating of the crane should be established as the rated load capacity, and the design rating for the degradable portion of the handling system should be identified to obtain the margin available for the maintenance program. The MCL should be plainly marked on each side of the crane for each hoisting unit. It is recommended that the critical-load-handling cranes should be continuously maintained above MCL capacity.

Response: An inspection procedure is currently in place to assure that the SFP crane is well maintained. The crane is a special purpose crane and is not capable of miscellaneous lifts. Therefore markings other than required by ANSI B30.16 are not necessary.

9.0 Operating Manual

Requirement: The crane designer and crane manufacturer should provide a manual of information and procedures for use in checking, testing, and operating the crane. The manual should also describe a preventive maintenance program based on the approved test results and information obtained during the testing. It should include such items as servicing, repair, and replacement requirements, visual examinations, inspections, checking, measurements, problem diagnosis, nondestructive examination, crane performance testing, and special instructions.

The operating instructions for all travel movements (vertical and horizontal movements or rotation, singly or in combination) incorporated in the design for permanent plant cranes should be clearly defined in the operating manual for hoisting and for trolley and bridge travel. The designer should establish the MCL rating and the margin for degradation of wear susceptible component parts.

Response: Vendor manuals were provided when the crane was purchased. The manuals contain information such as operation information, preventive maintenance, servicing, repair, and problem diagnosis. Procedures have been written to provide guidance on items such as testing and inspecting the crane, visual examinations, crane performance testing, and operating instructions.

10. Quality Assurance

Requirement: Although crane handling systems for critical loads are not required for the direct operation of a nuclear power plant, the nature of their function makes it necessary to ensure that the desired quality level is obtained. A quality assurance program should be established to the extent necessary to include the recommendations of this report for the design, fabrication, installation, testing, and operation of crane handling systems for safe handling of critical loads.

In addition to the quality assurance program established for site assembly, installation, and testing of the crane, applicable procurement documents should require the crane manufacturer to provide a quality assurance program consistent with the pertinent provisions of RG 1.28, "Quality Assurance Program Requirements (Design and Construction)," to the extent necessary.

The program should address all the recommendations in this report. Also included should be qualification requirements for crane operators.

Response: Quality assurance for the crane is established by the site. Modifications, tests, repairs, and inspections performed on the crane are performed in accordance with TVA QA requirements. Qualifications for crane operators are outlined in TVA procedures.

RAI Question 29:

Section 9.1.4.3.5, "Shipping Cask Integrity," of the SQN Final Safety Analysis Report describes that the radioactivity release from a fuel shipping cask drop event would be bounded by the release from the design-basis fuel handling accident. In Enclosure 4 to its letter of September 21, 2001, the TVA described that the loaded TPBAR shipping cask would be removed from the cask loading pit prior to completion of packaging for transportation. It is not clear that the radiological consequences from a dropped TPBAR shipping cask would be bounded by the evaluation of a fuel-handling accident involving a fuel assembly containing TPBARs. A review of the licensing basis for SQN indicates that the auxiliary building crane has not been designed to single-failure proof standards specified in NUREG-0554, "Single-Failure Proof Cranes for Nuclear Power Plants," and, therefore, shipping cask drops are credible design basis events.

In order to complete our review, the NRC staff requests that TVA provide either of the following evaluations:

a An evaluation of the maximum potential radiological consequences from a TPBAR shipping cask drop prior to sealing the cask and certifying it for shipment. This evaluation should consider the maximum lift height and maximum potential tritium release resulting from a drop of that height.

b *an evaluation comparing design features, operational controls, and analyses planned for implementation during TPBAR shipping cask lifts with those specified in the applicable section of NUREG-0612. This evaluation should address each specified item separately by describing what is planned for implementation and the basis for any difference in scope or depth relative to what is specified in NUREG-0612.*

Response:

SQN is currently in the process of converting the Main Hoist of the 125 Ton Capacity Auxiliary Building Crane to the Ederer X-Sam system. The Ederer X-Sam system has been accepted by the NRC as conforming to NUREG-0554 (Single Failure Proof Cranes for Nuclear Power Plants) requirements per Topical Report EDR-1, and will be utilized to lift the shipping casks.

RAI Question 30:

Although the change in spent fuel pool decay heat load resulting from irradiation of TPBARs is marginal, TVA has proposed a significant increase in the maximum spent fuel pool decay heat load. The additional decay heat load would result from fuel transfers to the spent fuel pool with shortened decay times. By utilizing margin in cooling capability associated with conservative values for component cooling water temperature and heat exchanger performance, the additional heat load does not result in an increase in spent fuel pool temperature. However, this change does significantly reduce the time-to-boil following a loss of spent fuel pool cooling and increase the maximum rate of coolant loss by evaporation. These changes reduce the overall reliability of evaporative cooling. Describe administrative controls that are or will be in place that ensure the reliability of the forced cooling system will be consistent with its importance to safety under high heat load conditions, such as the minimum required availability of forced cooling trains and associated support system trains (e.g., service water and component cooling water).

Response:

The normal configuration of the SFP cooling during refueling when the other unit is in power generation mode is to align SFP cooling with the operational unit, due to its lower overall heat loads. The unit in power operation mode requires that both trains of the CCS and ERCW systems be operable in accordance with TS requirements. In this alignment, SFP cooling is assured with two operable trains of cooling available.

During refueling operation modes with the other unit in either Hot Shutdown, Cold Shutdown, or Refueling, (and Power Generation if desired), the SFP is normally aligned to the unit with the least heat load. While a shutdown unit is not required to have two independent trains operable, administrative provisions are currently in place to "protect" or provide "defense in depth" to cooling trains associated with decay heat removal, via the RHR or the SFP CCS. This process is similar to provisions and controls during mid-loop RCS inventory operation. Additionally, the ability to place an increased heat load in the SFP can only be performed in accordance with procedures which require evaluating existing ERCW temperatures and determining achievable CCS temperatures for correlation with known SFP heat exchanger fouling rates to determine actual allowable SFP heat loads. This evaluation, as a part of overall outage management, assures that the plant evaluates and maintains decay heat removal systems for the short time period of the outage that maximum heat load is projected to be in the SFP.

ATTACHMENT 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
TECHNICAL SPECIFICATION (TS) CHANGE 00-06

RESPONSE TO QUESTION 18

WESTINGHOUSE ELECTRIC CORPORATION
(NON-PROPRIETARY)

RAI Question 18:

Please provide a complete description of the boric acid accumulation evaluation model that is used to establish compliance with Title 10, Code of Federal Regulations, Section 50.46(b)(5) and provide a complete assessment of model conservatisms and non-conservatisms. In addition, please compare your evaluation model prediction to your procedures for initiating hot-leg injection and assess conservatisms and non-conservatisms associated with the procedures.

Response:

The methodology used to confirm post-loss of coolant accident (LOCA) long-term core cooling capabilities for SQN establishes a post-LOCA hot leg switchover (HLSO) time to support realignment of the recirculation safety injection (SI) flow from the cold legs to the hot legs. This realignment is required to preclude boron precipitation in the reactor vessel following a large-break LOCA. For a cold leg break where injected SI water boils off due to decay heat, the potential exists for the boric acid solution in the reactor vessel to reach the boron precipitation point and block core cooling flow. The Westinghouse emergency core cooling system (ECCS) long term core cooling model confirms the existence of a coolable core geometry by establishing HLSO times which ensure that boron precipitation does not occur.

The HLSO analytical model used as part of the long-term core cooling methodology is based on the following assumptions:

1. A boric acid concentration level is computed over time for a core-region mixing volume. Other than the steam exiting through the hot legs and the corresponding makeup SI entering through the lower plenum, there are no assumed flow paths in or out of the mixing volume. All boric acid entering the mixing volume remains in the mixing volume prior to initiation of hot leg recirculation. The water/boric acid solution is well mixed in the mixing volume region. The water/boric acid solution in the vessel is assumed to be at atmospheric conditions, at a temperature of 212 degrees Fahrenheit. The collapsed mixture level of the core/upper plenum region is at the bottom of the hot leg flow area at the reactor vessel outlet nozzle. This level is the top of the mixing volume. The bottom of the mixing volume is at the level of the top of the lower core support plate. The lower plenum volume and barrel baffle region volume are not included in the mixing volume.

2. The boric acid concentration limit is the experimentally determined boric acid saturation concentration with an additional [ ]^{a,c} weight-percent margin factor. The calculation neglects any elevation of boiling temperature due

to concentration of boric acid in the core or due to backpressure from containment.

3. The decay heat generation rate is based on the 1971 American Nuclear Society (ANS) Standard for a finite operating time. The decay heat generation includes a core power multiplier to address instrumentation uncertainty as identified by Section I.A of Appendix K.

4. The boron concentration of the make-up SI is a calculated sump mixed mean boron concentration. The calculation of the sump mixed mean boron concentration assumes maximum mass and maximum boron concentrations for significant boron sources and minimum mass and maximum boron concentration for significant dilution sources.

5. Once realigned to hot leg recirculation, boron precipitation is precluded and core cooling is assured by established minimum recirculation flow criteria for the hot legs, cold legs, or simultaneous hot and cold leg injection.

The methodology described above is consistent with, or otherwise conservative with respect to, the methodology described in Reference 1. This methodology contains the following conservatisms.

1. All boric acid entering the mixing volume remains in the mixing volume. The simplified analytical model assumes that there are no paths for the boron or boric acid to leave the mixing volume. In fact, there are a number of paths for boric acid to leave the mixing volume. They include:

   - [

]^{a,c}

- [

  ]^{a,c}

- [

  ]^{a,c}

2. The assumed mixing volume is conservatively small. The simplified analytical model assumes a mixing volume that extends from the top of the lower core support plate to the bottom of the hot leg flow area at the reactor vessel outlet nozzle. A number of additional regions ([ ]^{a,c}) would see at least some mixing. Flow through these regions would result from thermal and density gradients throughout the core region.

3. The boric acid solubility limit is conservatively chosen. A [ ]^{a,c} weight-percent margin factor has been added to the experimentally determined boric acid saturation limit. The boric acid saturation limit is based on 14.7 pounds per square inch absolute (psia) conditions. The boric acid solubility limit increases significantly with pressure (i.e., the limit increases more than [ ]^{a,c} for each 10 pound per square inch increase in assumed pressure).

4. All heat removed from the core is assumed to be due only to the boiloff of saturated pure water at 14.7 psia. The simplified analytical model assumes that all heat removed from the core is due to the boil-off of pure water at 14.7 psia. In fact, there are a number of mechanisms for heat removal including:

   - [ ]^{a,c}

   - [ ]^{a,c}

- [ ]^{a,c}

- [ ]^{a,c}

The methodology also contains the following non-conservatisms.

1. The decay heat standard used in the Standard Westinghouse HLSO methodology uses a 1971 ANS Standard decay heat based on "finite" operation. This decay heat is calculated using [ ]^{a,c} core regions with [ ]^{a,c} hours operating time, respectively. This decay heat assumption is non-conservative in the following respects:

   - [ ]^{a,c}

   - [ ]^{a,c}

The SQN long-term cooling analysis prepared for the tritium production core used the standard Westinghouse evaluation methodology described above and established a required HLSO time of 5.59 hours post-LOCA (with [ ]^{a,c} weight-percent boron margin to the solubility limit). (It is significant to note that the boric acid accumulation model predicted that the actual boron precipitation point (i.e. no margin to the boric acid solubility limit) is not reached until 7.25 hours post-LOCA.) For additional conservatism and convenience, a revised HLSO time of 5.5 hours post-LOCA was planned for incorporation into the SQN emergency operating procedures for the tritium production core.

To address the effects of the analysis non-conservatism discussed above, an additional HLSO evaluation has been performed for SQN. This evaluation adds considerable conservatism to the original tritium production core long term cooling analysis and is considered non-standard with respect to the Westinghouse post-LOCA analysis methodology. Consistent with the standard evaluation methodology, the original Sequoyah tritium production core post-LOCA long-term cooling analysis used the 1971 ANS decay heat for finite operation without residual fissions and 0 percent (%) uncertainty. The evaluation performed revised this input and used the 10 CFR 50, Appendix K prescribed decay heat model (1971 ANS decay heat for infinite operation with 20% uncertainty). Other minor differences between the initial analysis and the evaluation include changes to the effective mixing volume and the calorimetric uncertainty assumptions. The effective mixing volume in the evaluation was conservatively calculated from the bottom of the active fuel to the bottom of the hot legs. The initial analysis included the lower fuel nozzle volume in the effective mixing volume. This additional volume was conservatively ignored in the subsequent evaluation. Since SQN has been approved for a 1.3% power measurement uncertainty recovery associated with the installation of a main feedwater leading edge flow measurement system, the core power in the evaluation was assumed to be 3455 mega-watt thermal (MWt) with a 1.007 calorimetric uncertainty factor. The initial analysis conservatively assumed a core power of 3455 MWt with the standard 1.02 calorimetric uncertainty factor. The small reduction in the effective mixing volume and the change to the total core power have only minor effects on the HLSO calculations.

Table 18-1 shows a comparison of the HLSO evaluation results to the original HLSO analysis results.

Table 18-1: Summary of the HLSO Evaluation Results

Case	HLSO Time	Margin to the 27.53 w/o precipitation limit
Analysis	7.25 hr	0.00 w/o
Analysis	5.59 hr	[ ]^{a,c} w/o
Appendix K Decay Heat Evaluation	5.35 hr	0.00 w/o
Appendix K Decay Heat Evaluation	4.15 hr	[ ]^{a,c} w/o
Appendix K Decay Heat Evaluation	3.00 hr	[ ]^{a,c} w/o

As seen in Table 18-1, using (a) the 10CFR50, Appendix K prescribed decay heat, (b) a decrease in effective mixing volume, and (c) a decrease in core power uncertainty, the SQN tritium production core HLSO time was established to be 4.15 hours post-LOCA with a [ ]^{a,c} weight-percent margin to the boron precipitation limit. This is compared to the original tritium production core HLSO analysis which predicted boron precipitation to occur at 5.59 hours with the same margin to the boron precipitation limit. With a further reduction of the HLSO time to 3 hours, the margin to the boron precipitation limit increased to [ ]^{a,c} weight-percent.

With the reduction in HLSO time, the ECCS performance at hot leg recirculation was also evaluated. All the minimum flow requirements were satisfied for a HLSO time of 3 hours. Since the core boil-off due to decay heat is greater at an earlier HLSO time, the minimum flow requirements are satisfied for a HLSO time ≥ 3 hours. (Note that the prescribed Appendix K decay heat model is used for ECCS recirculation performance calculations.)

Based on the results of the evaluation discussed above, the SQN emergency operating procedures will be revised to require initiation of hot leg ECCS recirculation 3 hours following a large break LOCA for the tritium production core rather than 5.5 hours. The 3-hour switchover time requirement does not increase operator burden during LOCA mitigation and recovery and will provide an added measure of conservatism with respect to the tritium production core long-term cooling analysis.

References

1. Letter from C. L. Caso (W) to T. M. Novak, (NRC), CLC-NS-309, "Long-term Boric Acid Buildup in the Core, Basic Phenomena," April 1, 1975.

2. WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information," April 1975 (WCAP-8472-A, Non-Proprietary Class 3).

3. WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model - Summary," June 1974.