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AUG 2 0 2001

TVA-WBN-TS-00-015 10 CFR 50.90

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

Gentlemen:

In the Matter of) Docket No(s). 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 1 - REVISION OF BORON CONCENTRATION LIMITS AND REACTOR CORE LIMITATIONS FOR TRITIUM PRODUCTION CORES (TPCs) - TECHNICAL SPECIFICATION (TS) CHANGE NO. TVA-WBN-TS-00-015

In accordance with the provisions of 10 CFR 50.90, TVA is submitting a request for an amendment to WBN's License NPF-90 to change the TS for Unit 1 to allow WBN to provide irradiation services for the U.S. Department of Energy (DOE). This change would allow WBN to insert Tritium Producing Burnable Absorber Rods (TPBARs) into the reactor core to support DOE in maintaining the nation's tritium inventory (Tritium Program). The proposed license amendment involves increasing the required boron concentration for both the cold leg accumulators (TS 3.5.1) and the refueling water storage tank (TS 3.5.4), removing the Region 2 burnup credit racks in the Spent Fuel Pool and clarifying fuel storage restrictions (TSs 3.7.15 and 4.3.3), adding a limit on the number of TPBARs that can be irradiated (TS Section 4.2.1), and the implementation of a TPBAR consolidation activity. This submittal also provides revisions to TS Bases to modify the switchover time for containment sump to hot leg recirculation (TS B3.5.2) and to modify the hydrogen recombiner section to properly describe the possible sources of hydrogen gas (TS B3.6.7).

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This proposed change is justified based on extensive analysis, testing, and evaluation of the TPBARs as reported previously by the DOE. DOE has previously submitted a classified/proprietary version (NDP-98-153, Revision 1) and an unclassified/non-proprietary version (NDP-98-181, Revision 1) of the Tritium Production Core (TPC) Topical Report for NRC review. NRC reviewed these TPC Topical Reports and issued NUREG-1672, "Safety Evaluation Report (SER) Related to the Department of Energy's Topical Report on the Tritium Production Core" documenting its review. TVA used both versions of the TPC Topical Report and the NRC SER in the preparation of this license amendment request and has completed the appropriate plant-specific evaluations and analyses recommended by these documents including the 17 interface items listed in NUREG-1672, Section 5.1. In order to maintain this license amendment request in an unclassified form, any classified text, tables, and figures that have been affected by the plant-specific application of TPBARs have been omitted from this submittal. Copies of the classified documents are available for NRC review at the Pacific Northwest National Laboratory (PNNL) offices.

TVA identified two issues that require further testing and analysis to confirm conservative assumptions. These issues involve lithium leaching and post LOCA material ejection from the TPBARs. Both issues incorporate current research and have been factored into the safety analyses enclosed. TVA has requested that DOE perform additional testing and analysis as described in Enclosure 4.

Enclosure 1 to this letter provides the description and evaluation of the proposed Technical Specification changes (Part A) and a description of the TPBAR consolidation activity (Part B) required for the Tritium Program. TVA requests NRC review under 10 CFR 50.90 to implement the changes necessary to irradiate TPBARs. This enclosure includes TVA's determination that the proposed changes do not involve a significant hazards consideration. In addition, an environmental impact consideration discussion is provided.

Enclosure 2 provides the appropriate TS pages marked to show the proposed changes. Enclosure 3 provides the revised TS pages.

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Enclosure 4 provides Westinghouse Report Number NDP-00-0344, Revision 1 which:

- contains information relative to items in the TPC Topical Report for which there is a WBN impact,

- contains confirmation of the plant specific confirming checks recommended by the TPC Topical Report,

- addresses the 17 plant-specific interface issues listed in NUREG-1672, section 5.1, and,

- addresses other items requested by the NUREG-1672 such as the TPBAR surveillance program, Lead Test Assembly (LTA) Post Irradiation results, and a discussion of proposed TS changes identified in the NUREG-1672 that are not required at WBN.

Four of the 17 interface items identified in NUREG-1672 affect Spent Fuel Pool (SFP) Cooling. To address these items, a change in the SFP cooling methodology was proposed in a letter to NRC dated April 10, 2001. The response to these four interface items and the methodology changes were addressed in that letter.

This submittal is consistent with pending technical specification changes contained in Steam Generator Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking submittal dated April 10, 2000 (plus updates), and one additional TS change planned in the near future for Dose Equivalent Iodine limits. However, updates for the Dose Equivalent Iodine limits have already been factored into the tritium analysis.

Portions of Enclosures 1 (TPBAR consolidation activity) and 4 were previously submitted on May 1, 2001. In that submittal, areas labeled as "Information to be provided later," were identified. This submittal provides that information.

The WBN Plant Operations Review Committee and the WBN Nuclear Safety Review Board have reviewed these proposed changes and have determined that operation of WBN Unit 1 in accordance with these proposed changes will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

In order to meet DOE's Tritium program requirements, TVA requests that this amendment be approved within one year of this submittal and the revised Technical Specifications be made effective during the refueling outage when the TPBARs are planned to be inserted.

There are no new regulatory commitments being made by this submittal. If you have any questions about this license amendment request, please contact P. L. Pace at (423) 365-1824.

Sincerely,

W. R. Lagergren

W. R. Lagergren

Enclosures
cc: See page 5

Subscribed and sworn to before me
on this 20th day of August 2001

E. Jeannette Long
Notary Public

My Commission Expires May 21, 2005

cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1
DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-015
AND TPBAR CONSOLIDATION ACTIVITY
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

PART A - PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-015

I. DESCRIPTION OF THE PROPOSED CHANGE

In order to irradiate Tritium Producing Burnable Absorber Rods (TPBARs) at WBN, changes to the TSs and the associated TS Bases discussions need to be made. The first two changes involve TSs 3.5.1 and 3.5.4 which will require increasing the boron concentration in both Cold Leg Accumulators (CLAs) and Refueling Water Storage Tank (RWST) which stem from fuel core design. The RWST change also involves modifying the associated TS Bases section B3.5.4. The third and fifth changes which involve TSs 3.7.15 (and associated TS Bases pages) and 4.3.3 respectively, delete the Region 2 burnup credit rack specifications and more fully describe storage restrictions based on burnup. The fourth change is to TS 4.2.1 which involves incorporating into the Design Features Section 4.0 the maximum number of TPBARs that can be inserted into the reactor core in an operating cycle. The sixth and seventh changes, respectively, are revisions to TS Bases B3.5.2 to reduce the switchover time for containment sump to hot leg recirculation from 9 hours to 5.5 hours and to the TS Bases B3.6.7 discussion involving the hydrogen recombiners to properly describe the possible sources of hydrogen gas. Each of these changes are described below:

A. TS 3.5.1 - Cold Leg Accumulator - Boron Concentration Increase

This change is requested to Surveillance Requirement (SR) 3.5.1.4 to increase the Cold Leg Accumulator Boron Concentration from the present range of 2400 to 2700 ppm to a range of 3500 to 3800 ppm.

B. TS 3.5.4 and associated TS Bases Page - Refueling Water Storage Tank - Boron Concentration Increase

This change is requested to SR 3.5.4.3 to increase the Refueling Water Storage Tank Boron Concentration from the present range of 2500 to 2700 ppm to a range of 3600 to 3800 ppm.

In addition, changes are requested to the associated TS bases section B3.5.4 entitled, "Applicable Safety Analyses," to increase the minimum RWST boron concentration from 2500 ppm to 3600 ppm to reflect the minimum value used in the post Loss of Coolant Accident sump analysis for core subcriticality and to increase the maximum RWST boron concentration from 2700 ppm to 3800 ppm to reflect the maximum values used in the hot leg switchover time calculation.

C. TS 3.7.15 and the associated TS Bases Pages - Plant Systems/Spent Fuel Assembly Storage

This change is requested to remove reference to Figure 3.7.15-1 and to remove Figure 3.7.15-1 from this section which results in the removal from use of the Region 2 burnup credit racks. As a result, the necessary text changes to the associated TS Bases Pages are being made to reflect the above described removal.

D. TS 4.2.1 - Design Features/Reactor Core/Fuel Assemblies

A change is requested to section 4.0, Design Features, to allow the insertion of a maximum of 2304 TPBARs into the WBN reactor core for irradiation purposes. The specific number of TPBARs to be irradiated during a given cycle would be identified in the Reload Safety Evaluation Report but will, in all cases, be less than or equal to 2304.

Currently, in paragraph 4.2.1 on TS Page 4.0-1, the last sentence reads as follows:

> For Unit 1, Cycle 2, Watts Bar is authorized to place a limited number of Tritium Producing Burnable Absorber Rod lead test assemblies into the reactor in accordance with TVA's application dated April 30, as supplemented June 18, July 21 (3 letters), and August 7 and 21, 1997.

This request would change this sentence to read as follows:

> For Unit 1, Watts Bar is authorized to place a maximum of 2304 Tritium Producing Burnable Absorber Rods into the reactor core in an operating cycle.

E. TS 4.3.3 - Design Features/Fuel Storage/Capacity

This change is requested to modify section 4.3.3 to remove paragraph 4.3.3.2 resulting in the reduction of the spent fuel pool storage capacity from 1610 to 1386 fuel assemblies due to removal from TS of the Region 2 burnup credit racks. It should be noted that these racks had not yet been installed in the WBN Spent Fuel Pool. These changes also provides a more detailed description of storage restrictions based on burnup.

F. Bases 3.5.2 - Emergency Core Cooling Systems/ECCS - Operating

This change is being made to revise the switchover time for containment sump to hot leg recirculation from 9.0 hours to 5.5 hours.

G. Bases 3.6.7 - Hydrogen Recombiners

As a result of the tritium program, a change is being made to the TS Bases for hydrogen recombiners to include tritium inside the TPBARs as a possible source. This change would modify items a. and c. on page B 3.6-44 to read as follows:

a. A metal steam reaction between the zirconium fuel rod cladding, TPBAR zirconium internals, and the reactor coolant;

c. Hydrogen in the RCS at the time of LOCA (i.e., hydrogen dissolved in the reactor coolant, hydrogen gas in pressurizer vapor space, and tritium contained in TPBARS); or

II. <u>REASON FOR THE PROPOSED CHANGE</u>

A. TS 3.5.1 - Cold Leg Accumulator - Boron Concentration Increase

The post-LOCA long term core cooling analysis requires maintaining a subcritical boron concentration following a LOCA after all boration sources are injected and mixed in the containment sump. These boration sources include the CLAs, the RWST, and the melted ice from the ice condenser.

When large amounts of excess neutron poison are added to a core, such as with TPBARs, there is competition for neutrons from all the poisons and the negative worth of each poison (including the reactor coolant system (RCS) boron) decreases. The positive reactivity insertion due to the negative moderator coefficient that occurs during the cooldown from hot full power to cold conditions following the LOCA must be entirely overcome by RCS boron. Because the RCS boron is now worth less, it takes a higher concentration to maintain subcriticality. The ice (at approximately 1900 ppm) is a dilution source which has to be overcome by the RWST concentration to reach a mixed sump concentration high enough to prevent criticality.

Therefore, the CLAs boron concentration will have to be increased to the values requested in Section I.A.

B. TS 3.5.4 and the associated TS Bases Page - RWST Boron Concentration Increase

Based on the discussion in Item A, the RWST boron concentration will also have to be increased to the values requested in Section I.B.

C. TS 3.7.15 and the associated TS Bases Pages - Plant Systems/Spent Fuel Assembly Storage

The purpose for removing Figure 3.7.15-1, and the reference to it, is that TVA has determined, since the time that Region 2 burnup credit racks were licensed, that it does not plan to install or utilize this storage option as described by TS section 4.3.3 which is being changed below. As a result, the necessary text changes to the associated TS Bases are being made to reflect the above described removal.

D. TS 4.2.1 - Design Features/Reactor Core/Fuel Assemblies

The purpose for this change is to place a limit on the number of TPBARs that can be inserted into the reactor core in an operating cycle based on plant safety analyses. The specific number of TPBARS to be irradiated during a given cycle would be identified in the Reload Safety Evaluation Report, but will not be greater than 2304.

E. TS 4.3.3 - Design Features/Fuel Storage/Capacity

The purpose for removing section 4.3.3.2, resulting in the reduction of the total spent fuel pool storage capacity, is that TVA has determined, since the time that Region 2 burnup credit racks were licensed, that it does not plan to install or utilize this storage option as described by section 4.3.3.2.

The purpose of providing a more detailed description of fuel storage restrictions is to allow more flexibility in the storage of spent fuel.

F. Bases 3.5.2 - Emergency Core Cooling Systems/ECCS - Operating

The purpose for revising the time for the containment sump switchover to hot leg recirculation is due to the increase, as stated above, in the maximum RWST and accumulator boron concentrations. Since the initial mixed boron concentrations are higher and the precipitation concentration is reached sooner, the hot leg switchover value of 9 hours is being changed to 5.5 hours.

G. Bases 3.6.7 - Hydrogen Recombiners

The purpose for modifying items a. and c. in the hydrogen recombiner discussions is to include tritium and hydrogen inventories existing in the TPBARs that would be available for release during postulated accidents. This revision will properly describe the sources that have been considered in evaluating the adequacy of the combustible gas control functions.

III. SAFETY ANALYSIS

A. TS 3.5.1 - Cold Leg Accumulator - Boron Concentration Increase

1. LOCA Related Analyses

a. Large Break LOCA (LBLOCA)

Accumulator boron concentration is used in the Best Estimate LOCA point kinetics model to maintain subcriticality during the reflood period of the transient. During the refill period of the LBLOCA transient, the water in the reactor vessel is almost entirely from the accumulators since most, if not all, RCS inventory has either discharged out the break or has flashed to steam. After this, during the reflood period, make-up is from the RWST. Because the mixed fluid from both the accumulator and RWST sources has experienced minimal dilution from the RCS, the final concentration would be somewhere in the range between 3500 and 3600 ppm. The analysis in support of the Post LOCA Long Term Core Cooling requirements (see item d, below) demonstrates that the core remains subcritical with a mixed sump boron concentration which is less than the 3500 to 3600 ppm range. This demonstrates that the core will remain subcritical during the transient as well as after. As such, it is concluded that the proposed minimum concentrations of 3500 ppm for the accumulators and 3600 ppm for the RWST will be acceptable for the Watts Bar TPC design from a Best Estimate LOCA standpoint. In addition, there is no increase in the Best Estimate LBLOCA PCT; therefore, there continues to be a high level of probability that the ECCS acceptance criteria limit is not exceeded with regard to the LBLOCA analysis. Thus, the increase in the accumulator and RWST boron concentrations would have no adverse effect on the Best Estimate LBLOCA analysis results.

b. Small Break LOCA (SBLOCA)

The SBLOCA analysis does not take credit for the boron present in the RWST and the accumulators. The calculated PCT is not a function of the boron concentration level in the core. Thus, an increase in the accumulator and RWST boron concentrations would have no adverse effect on the SBLOCA analysis results.

c. Reactor Vessel Blowdown and Loop Forces

The LOCA blowdown hydraulic loads occur within the first few seconds of the LOCA transient and thus are not a function of the boron concentration level in the accumulators or RWST. Thus, an increase in the boron concentration levels in the accumulators and RWST would have no effect on the LOCA hydraulic forces calculation.

d. Post LOCA Long Term Core Cooling Requirements

The licensing basis commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA. Minimum boron concentrations are assumed in the calculation for each borated water source. For this calculation, the minimum RWST boron concentration is 3600 ppm and the minimum accumulator concentration is 3500 ppm.

Testing has indicated that TPBARs can experience cladding breach at LBLOCA conditions if the cladding temperature and internal pressure of the TPBARs reach limiting values. Consequently, the post-LOCA critical boron calculations accounted for the potential loss of a $LiAlO_2$ pencil, as well as partial leaching of lithium from the remaining pencils. Based on conservative assumptions, the calculations confirm that the tritium production core will remain subcritical following a LOCA.

e. Hot Leg Switchover Time to Prevent Boron Precipitation

The hot leg recirculation switchover time is determined for inclusion in emergency procedures to preclude boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level and on the RCS, RWST, accumulator and other (i.e., ice melt) water volumes and boron concentrations. In the event of a cold leg break during which the ECCS is aligned to the RCS cold legs, boron concentration in the core region increases due to boil-off of water. To preclude boron precipitation, the ECCS is realigned to the RCS hot legs at the hot leg switchover time.

The increase in the maximum RWST and accumulator boron concentrations results in a reduction in the hot leg switchover time because initial mixed boron concentrations are higher, and the precipitation concentration is reached sooner. The current hot leg switchover value of 9 hours will be changed to 5.5 hours.

2. Non-LOCA Transient Analysis

The only non-LOCA event that assumes accumulator actuation is the Major Rupture of a Main Steamline event. This event, however, assumes delivery of the minimum amount of boron to the core to maximize the expected return to power. Therefore, the current licensing basis main steamline break analysis bounds the proposed conditions and the results and conclusion presented in the Updated Final Updated Safety Analysis Report (UFSAR) remain valid. The TPBAR core design has not changed any bounding value assumed for the key safety analysis parameters used in the analysis of this event.

The only non-LOCA event that assumes a maximum RWST boron concentration is the Inadvertent Operation of ECCS event. Two separate cases are considered for this event. The first case is investigated to verify that the departure from nucleate boiling ratio (DNBR) safety limits are not violated, and the second case examines the potential for pressurizer filling. For the DNBR case, an increase in the boron concentration results in a decrease in reactor power, hence a decrease in coolant temperature and pressure. The decrease in reactor power and coolant temperature result in an increase to the DNBR, while a decrease in pressure results in a decrease in DNBR. Thus, the opposing DNB trends offset each other, resulting in no impact to the DNB case due to the increase in boron concentration. With respect to the pressurizer fill case, reactor trip is assumed to occur at event initiation and core boron concentration does not impact post-trip decay heat generation, resulting in no impact on pressurizer filling results. The TPBAR core design has not changed any bounding value assumed for the key safety analysis parameters used in the analysis of this event.

3. Steamline Break (SLB) Mass and Energy (M&E) Releases

   The SLB M&E analyses are performed for the containment integrity evaluation, compartment pressurization analysis and equipment qualification. These analyses assume the minimum allowable boron concentrations for the RWST and accumulators to minimize the amount of boron delivered to the core. The control rods provide the safety analysis value for the shutdown margin for this event. Therefore, the proposed boron concentration increase has no adverse impact. The TPBAR core design has not changed any bounding value assumed for the key safety analysis parameters used in the analysis of this event.

4. Steam Generator Tube Rupture (SGTR)

   During the SGTR event, a low pressurizer pressure signal actuates the SI system which delivers flow from the RWST to the RCS. The borated water from the RWST helps to maintain the reactor in a shutdown condition after the tube rupture has occurred. The increase in the RWST concentration will lead to a higher boration rate and ultimately increase the overall RCS boron concentration. The accumulators are not modeled in the event since the RCS pressure remains above the accumulator injection pressure. Reload tritium production cores will be evaluated to demonstrate adequate shutdown margin for this event. The TPBAR core design has not changed any bounding value assumed for the key safety analysis parameters used in the analysis of this event.

5. Containment Mass and Energy Releases

   The LOCA temperature and pressure response analyses which are performed for containment integrity, compartment evaluation, and equipment qualification do not model the RWST and accumulator boron concentrations. Thus, the changes in concentration do not affect these analyses.

6. Nuclear Steam Supply System (NSSS) Systems and Components

   a. Mechanical Components and Systems

      The impact of an increase in the boron concentration range in the RWST and accumulators was assessed with respect to the mechanical and fluid system components. This increase in concentration will cause a decrease in the pH of the liquid and therefore required a review regarding the integrity of the RWST and accumulator materials, as well as other RCS component materials. This evaluation demonstrates that the integrity and operability of potentially affected equipment and systems will be maintained.

The RWST provides borated water to the refueling canal, charging pumps, SI pumps, containment spray pumps, and residual heat removal pumps. The accumulators supply water to the RCS during certain accident conditions. The immediate effect of raising the boric acid concentration in the RWST to 3800 ppm will be a decrease in the pH of the liquid. To assess the magnitude of this decrease, pH values of boric acid solutions containing 2700, 3250, and 3800 ppm at 40°F, 77°F, and 125°F were computed. These values are listed in the table below. The lowest and highest temperatures chosen, 40°F and 125°F, bound the range the RWST is expected to experience while 77°F is the temperature which the RWST liquid is expected to exhibit most of the time.

Table
pH of Boric Acid Solutions

Boron (ppm)	pH at 40 °F	pH at 77 °F	pH at 125 °F
2700	4.39	4.39	4.43
3250	4.27	4.28	4.32
3800	4.17	4.18	4.22

An inspection of the above table confirms that the pH of the RWST and accumulator liquids decreases very slightly when the boron concentration is increased from 2700 ppm to 3800 ppm. Specifically, the maximum reduction in pH in going from 2700 to 3800 ppm is only 0.22. This minimal pH decrease is not expected to cause new concerns regarding the integrity of the RWST or accumulator material or any other stainless steel surfaces that may come in contact with the RWST and accumulator liquids in the above temperature range.

In addition, structural carbon steel surfaces in containment during either the injection or recirculation phase following a postulated LOCA are protected by approved coatings against corrosion. Wherever there are unprotected carbon steel surfaces, some corrosion is expected to take place in the moist air of the containment. The unprotected surfaces will receive a spray of RWST liquid containing 3800 ppm boron during the containment spray injection phase following a LOCA, but the slightly lower pH of the spray will not have a measurable effect on the corrosion rate of carbon steel. Based on engineering judgement, the slight pH decrease of the RWST and accumulator liquids resulting from the proposed increase in boron concentration to 3800 ppm will not cause any new corrosion concerns to unprotected (unpainted) carbon steel surfaces in the containment. During the recirculation phase following a LOCA, the expected pH of the containment sump is such that no significant corrosion of in-containment carbon steel surfaces is expected.

Finally, the solubility of boric acid at 40°F, 77°F, and 125°F is about 5402 ppm, 9493 ppm, and 18,758 ppm, respectively. Therefore, a boron concentration of 3800 ppm will remain in solution at the temperatures the liquids in the Watt Bar Unit 1 RWST and accumulators may experience.

b. Instrumentation and Control Systems

An increase in boron concentration can impact accident/post-accident chemistry conditions in the containment building. With respect to the environmental qualification (EQ) of Class 1E equipment, such changes are only significant if the final pH of the containment sump solution differs greatly from that simulated during qualification testing. The intended objective is:

- to achieve and maintain pH above neutral (7.0) to preclude the possibility of chloride induced stress corrosion cracking, and

- to maintain a reasonable upper limit on pH (10.5 - 11.0) such that there is no significant degradation of polymer materials in the presence of strong alkali solutions.

Chloride induced stress corrosion cracking is a concern applicable to any stainless steel equipment located in the containment, but not unique to Class 1E equipment. Upper limits on pH range are established to provide adequate margin above the minimum pH (neutral 7.0) and with consideration of the likely non-metals used as vital sealing components of equipment. In practice, it is the non-metals that are selected for their endurance in the presence of the upper pH level selected by the equipment designer.

The purpose of chemistry conditions during EQ testing is to simulate a reasonable upper pH limit. The typical upper range limit value is 10.5 to 10.7 pH (varies among the specific tests performed). The intent is to affirm that chemistry, in conjunction with the extremes of pressure and temperature, does not result in a common mode failure of critical equipment/components. This is also the typical practice of other qualifiers of Class 1E equipment in that the choice of specific pH values simulated during testing will vary. TVA's qualification program for 10 CFR 50.49 equipment addresses the chemistry in determination of the qualification for use inside containment.

A calculation of the post LOCA sump pH with the higher boron concentrations indicates that the minimum long term sump pH will be reduced from a range of 8.0 to 10.5 to a new range of 7.5 to 10.0. The minimum pH value of 7.5 pH will not result in an adverse impact to the qualification of Class 1E equipment or its components. There is no impact to the qualification of Class 1E equipment.

c. Emergency Operating Procedures (EOPs)

TVA will revise the EOPs to reflect the new hot leg switchover time defined previously in Section III.A.1.e of this submittal.

d. Radiological Dose and Hydrogen Production

The increase in RWST and accumulator boron concentrations and subsequent slight decrease in containment sump and spray pH does not impact the LOCA dose evaluation. While higher pH helps maintain volatile iodine in solution and lower pH drives the equilibrium to favor volatile iodine in a gaseous state, the change in sump pH is not sufficient to result in any measurable change in post LOCA releases. Furthermore, current radiological analyses do not take credit for iodine removal efficiencies based on sump pH.

The analysis for iodine removal assumes that the ice condenser is the primary removal mechanism and no credit is taken for Iodine removal by containment spray. Since there is no change in the concentration of the sodium tetraborate in the ice, the existing analysis for iodine removal is still valid. Iodine solubility has been correlated with alkaline aqueous solutions. The pH of the containment sump and spray remains basic and there is no impact on the solubility of iodine in the sump and core fluid. Therefore, the proposed change in RWST and accumulator boron concentration will not affect the LOCA radiological dose calculations and the present analysis remains bounding.

The slight decrease in sump, core and spray fluid pH has been evaluated to not significantly impact the corrosion rate (and subsequent generation of Hydrogen) of Aluminum and Zinc inside containment so that the present analysis remains bounding. In addition, the decreased sump, core and spray fluid pH will not affect the amount of hydrogen generated from the radiolytic decomposition of the sump and core solution.

B. TS 3.5.4 and the associated TS Bases Page - RWST - Boron Concentration Increase

The evaluation for the previous section also applies for the RWST.

C. TS 3.7.15 and the associated TS Bases Pages - Plant Systems/Spent Fuel Assembly Storage

The Region 2 burnup credit racks described in TS section 4.3.3 are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Therefore, since they are not installed, there is no safety impact due to this change.

D. TS 4.2.1 - Design Features/Reactor Core/Fuel Assemblies

This proposed change is justified based on extensive analysis, testing, and evaluation of the TPBARs as reported previously in the TPC Topical Report and the evaluations performed for WBN described in the Westinghouse Topical Report NDP-00-0344. TVA has performed the confirming checks recommended by the DOE TPC Topical Report and plant specific evaluations requested by the NRC NUREG 1672.

TVA has reviewed these changes and has identified two issues that required further testing and analysis. These issues are lithium leaching from the TPBAR failure during operation and post LOCA material ejection from the TPBARs. See Sections 2 and 3 of Enclosure 4. Both issues incorporate current research and have been factored into the safety analyses enclosed. However, TVA has requested that DOE perform additional confirmatory testing as described in Enclosure 4. Details of these additional evaluations, confirming checks, and analyses to support the conclusion of safe operation can be found in Enclosure 4 of this submittal.

E. TS 4.3.3 - Design Features/Fuel Storage/Capacity

The Region 2 burnup credit racks described in this TS section are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Due to the deletion of the Region 2 racks, the additional detail provided clarifies existing storage restrictions. Therefore, since they are not installed, there is no safety impact due to this change.

F. Bases 3.5.2 - Emergency Core Cooling Systems/ECCS - Operating

Due to the increases of the boron concentrations in the RWST and the accumulators, initial mixed boron concentrations are higher and the precipitation concentration is reached sooner. Therefore, as a result, the hot leg switchover value of 9 hours is being changed to 5.5 hours.

G. Bases 3.6.7 - Hydrogen Recombiners

The modification to items a. and c. to include tritium from TPBARS in the Bases only serves to completely describe considerations included in the evaluation for TPBAR irradiation. This change does not alter the TS requirements or the functions for the hydrogen recombiner at WBN. This is an administrative addition for completeness and accuracy and will not impact nuclear safety. Details on the potential amount of hydrogen added by the TPBARs and the effect on the hydrogen recombiner functions can be found in Enclosure 4 of this submittal.

PART B - TRITIUM PRODUCING BURNABLE ABSORBER RODS (TPBARs) CONSOLIDATION ACTIVITY

I. DESCRIPTION OF THE PROPOSED CHANGE

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

Consolidation Sequence:

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

Approximately 30 days after refueling is complete, TPBAR consolidation begins. The canisters (See Enclosure 4, Figure 1.5.1-3) to receive the irradiated TPBARs are transferred into the spent fuel pool, and placed into the consolidation fixture when required. A TPBAR Assembly is then withdrawn from its storage location and moved from the spent fuel pool to the consolidation fixture using the TPBAR Assembly handling tool suspended from the Spent Fuel Pit (SFP) Bridge Crane. A TPBAR release tool is then utilized by personnel on the platform to detach individual TPBARs from the baseplate. The TPBAR slides along frame guides, through a funnel and into a roller brake, to limit its' velocity, and then into the consolidation canister. The funnel, roller brake assembly, and canister are angled at approximately 15° to enable the TPBARs to stack efficiently into the canister to maximize the loading. Activities take place underwater at a safe shielding water depth.

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

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

II. REASON FOR THE PROPOSED CHANGE

Equipment and methodologies do not currently exist for TPBAR consolidation and preparation for shipment. TVA requests NRC review under 10 CFR 50.90 to implement the changes necessary to irradiate TPBARs.

III. SAFETY ANALYSIS

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

1. Seismic Qualification Of The SFP Racks With Loaded Consolidation Canisters

   The Spent fuel pool racks have been seismically qualified containing Consolidation Canisters loaded with up to 300 TPBARs and have been found acceptable.

2. Heat Produced By The Irradiated TPBARs In A Consolidation Canister

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

3. Maintaining Criticality Limits For The Spent Fuel Racks Containing Loaded Canisters

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

4. Fuel Handling And Storage For Assemblies Containing TPBARs

The weight of a fuel assembly with 24 TPBARs and it's hold-down assembly (62 additional lbs. for TPBARs) is less than an assembly with a Rod Control Cluster (94 additional lbs.), and therefore is bounded by the current assumed weight of assembly for purposes of analyzing fuel handling and storage facilities. The TPBAR equipped fuel assembly has the same external configuration to interface with the fuel handling/storage equipment. Additionally, this weight is conservative for purposes of defining a NUREG-0612 "Heavy Load."

5. TPBAR Assembly Handling For Consolidation

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

6. TPBAR Consolidation Canister Handling

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

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

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

      a. Since the SFP Bridge Crane has a capacity of 4000 lbs. and the weight of the submerged loaded canister is approximately 700 lbs., the crane has safety factors twice the normally required values.

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

   2. The lifting tool is provided with a safety lanyard attached to a hoist trolley to limit canister descent in the fuel pool to such an extent that spilling of the TPBARs out of the open topped canister is prevented. The lanyard is sized to stop the canister from a maximum hook speed of 40 feet per minute. Administrative requirements require that the safety lanyard be attached to the lifting tool when the canister is not engaged in a SFP rack cell, the consolidation fixture holster, or cask by at least 12".

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

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

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

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

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

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

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

B. The lifting devices are designed to the requirements of ANSI N14.6 for Critical Loads with increased safety factors and load test weights, in addition to the design, fabrication, inspection, and testing contained in Sections 1 through 6 of ANSI N14.6, thereby rendering it equivalent single-failure-proof.

8. TPBAR Transport Cask Handling

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

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

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

   2. The lifting device is designed to the requirements of ANSI N14.6 for Critical Loads with increased safety factors and load test weights, in addition to the design, fabrication, inspection, and testing contained in Sections 1 through 6 of ANSI N14.6, thereby rendering it equivalent single-failure-proof.

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

9. Worker Radiation Exposure During TPBAR Consolidation Activities

The TPBAR handling and consolidation equipment is designed and configured such that minimum water shielding in the Spent Fuel Pool and Cask Loading Pit is maintained to keep dose rates ALARA. Tool design/features prevent inadvertently raising the TPBAR assemblies, loaded canisters or post consolidation baseplates above safe shielding depths.

Personnel will work on a platform 24″ above SFP normal water level over the deep end of the Cask Loading Pit. The platform is designed to accommodate lead shielding, if required, for personnel protection.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In order to irradiate Tritium Producing Burnable Absorber Rods (TPBARs) at Watts Bar Nuclear Plant (WBN), changes to the plant Technical Specifications (TSs) and the associated TS Bases discussions need to be made. The first two changes involve TSs 3.5.1 and 3.5.4 which will require increasing the boron concentration in both Cold Leg Accumulators (CLAs) and Refueling Water Storage Tank (RWST) which stem from fuel core design. The RWST change also involves modifying the associated TS Bases section B3.5.4. The third and fifth changes which involve TSs 3.7.15 (and associated TS Bases Pages) and 4.3.3 respectively, delete the Region 2 burnup credit rack specifications and more fully describe storage restrictions based on burnup. The fourth change is to TS 4.2.1 which involves incorporating into the Design Features Section 4.0 the maximum number of TPBARs that can be inserted into the reactor core in an operating cycle. The sixth and seventh changes are revisions to TS Bases B3.5.2 to reduce the switchover time for containment sump and to the TS Bases B3.6.7 discussion involving the hydrogen recombiners to properly describe the possible sources of hydrogen gas. The final change involves the addition of a TPBAR consolidation activity.

TVA has concluded that operation of Watts Bar Nuclear Plant (WBN) Unit 1 in accordance with the proposed changes to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

1. TS 3.5.1 - Cold Leg Accumulator - Boron Concentration Increase

   The accumulator boron concentration does not affect any initiating event for accidents currently evaluated in the Updated Final Safety Analysis Report (UFSAR). The increased concentrations will not adversely affect the performance of any system or component which is placed in contact with the accumulator water. The integrity and operability of the stainless steel surfaces in the accumulator and affected Nuclear Steam Supply System (NSSS) components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials. The design, material and construction standards of all components which are placed in contact with the accumulator water remain unaffected. Therefore, the possibility of an accident has not been increased.

The consequences of an accident previously evaluated in the UFSAR will not be increased. The change in the concentrations increase the amount of boron in the sump during a Loss of Coolant Accident (LOCA). The increased boron in the sump is sufficient to maintain the core in a subcritical condition. Testing has indicated that TPBARs can experience cladding breach at Large Break LOCA (LBLOCA) conditions if the cladding temperature and internal pressure of the TPBARs reach limiting values. Consequently, the post-LOCA critical boron calculations accounted for the potential loss of a $LiAlO_2$ pencil, as well as partial leaching of lithium from the remaining pencils. Based on conservative assumptions, the calculations confirm that the tritium production core will remain subcritical following a LOCA. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant Emergency Operating Procedures (EOPs). Thus, there will be no boron precipitation in the core following a LBLOCA.

The only non-LOCA event that assumes accumulator actuation is the Major Rupture of a Main Steamline event, however, it assumes a minimum amount of boron. Furthermore, there is no impact on the SGTR event since the accumulators are not assumed to be actuated, and the SLB M&E release evaluation relies on control rods for shutdown margin and assumes a minimum boron concentration.

In addition, the increase in accumulator boron concentrations and subsequent slight decrease in containment sump and spray pH does not impact the LOCA dose evaluation since the analysis of record does not credit sump pH as an input or assumption regarding volatile iodine removal efficiencies. Therefore, the present analysis remains bounding. Also, the slight decrease in sump, core and spray fluid pH has been evaluated to not significantly impact the corrosion rate (and subsequent generation of Hydrogen) of Aluminum and Zinc inside containment. Further, the decreased sump, core and spray fluid pH has been evaluated to not affect the amount of hydrogen generated from the post-LOCA radiolytic decomposition of the sump and core solution. The likelihood of containment failure due to hydrogen deflagration is therefore not impacted by pH changes.

In view of the preceding, it is concluded that the proposed change will not increase the radiological consequences of an accident previously evaluated in the FSAR.

2. TS 3.5.4 and the associated TS Bases Page - Refueling Water Storage Tank (RWST) - Boron Concentration Increase

   The RWST boron concentration does not affect any initiating event for accidents currently evaluated in the UFSAR. The increased concentration will not adversely affect the performance of any system or component which is placed in contact with the RWST water. The integrity and operability of the stainless steel surfaces in the RWST and affected NSSS components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials. The design, material and construction standards of all components which are placed in contact with the RWST water remain unaffected. Therefore, the probability of an accident has not changed.

   The consequences of an accident previously evaluated in the UFSAR will not be increased. The change in the concentrations increases the amount of boron in the sump following a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition. This analysis assumes partial leaching. Testing has indicated that TPBARs can experience cladding breach at LBLOCA conditions if the cladding temperature and internal pressure of the TPBARs reach limiting values. Consequently, the post-LOCA critical boron calculations accounted for the potential loss of a $LiAlO_2$ pencil, as well as partial leaching of lithium from the remaining pencils. Based on conservative assumptions, the calculations confirm that the tritium production core will remain subcritical following a LOCA.

Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs. Thus, there will be no boron precipitation in the core following a LOCA.

The Inadvertent Operation of Emergency Core Cooling System (ECCS) event is the only non-LOCA event which assumes the maximum RWST boron concentration, and an evaluation has shown that the proposed increase does not cause an adverse impact on this transient.

The Steam Line Break (SLB) mass and energy (M&E) release evaluation relies on control rods for shutdown margin and assumes a minimum boron concentration. For the Steam Generator Tube Rupture (SGTR) event, the increased boron concentration will help maintain adequate shutdown margin, which will be evaluated as part of the reload process.

In addition, the increase in RWST boron concentrations and subsequent slight decrease in containment sump and spray pH does not impact the LOCA dose evaluation. While higher pH helps maintain volatile iodine in solution and lower pH drives the equilibrium to favor volatile iodine in a gaseous state, the change in sump pH is not sufficient to result in any measurable change in post LOCA releases.

Furthermore, current radiological analyses do not take credit for volatile iodine removal efficiencies based on sump pH. Therefore, since the change in pH is minimal, and no credit is taken in release analysis, the present analysis remains bounding. Also, the slight decrease in sump, core and spray fluid pH has been evaluated to not significantly impact the corrosion rate (and subsequent generation of Hydrogen) of Aluminum and Zinc inside containment and the present analysis remains bounding. Further, the decreased sump, core and spray fluid pH has been evaluated to not affect the amount of hydrogen generated from the radiolytic decomposition of the sump and core solution and therefore will not challenge containment integrity.

In view of the preceding, it is concluded that the proposed change will not increase the consequences of an accident previously evaluated in the FSAR.

3. TS 3.7.15 and the associated TS Bases Pages - Plant Systems/Spent Fuel Assembly Storage

   The Region 2 burnup credit racks described in TS section 4.3.3 are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Therefore, since they are not installed, there is no increase in the probability or consequences of an accident previously evaluated.

4. TS 4.2.1 - Design Features/Reactor Core/Fuel Assemblies

   The insertion of TPBARs into the WBN reactor core does not adversely affect reactor neutronic or thermal-hydraulic performance; therefore, they do not significantly increase the probability of accidents or equipment malfunctions while in the reactor. The neutronic behavior of the TPBARS mimics that of standard burnable absorbers with only slight differences which are accommodated in the core design. The reload safety analysis performed for WBN Unit 1 prior to each refueling cycle will confirm that any minor effects of TPBARS on the reload core will be within fuel design limits.

   As described in the TPC Topical, the TPBAR design is robust to all accident conditions except the large break LOCA where the rods are susceptible to failure. However, the failure of TPBARs has been determined to have an insignificant effect on the thermal hydraulic response of the core to this event, and analysis has shown that the core will remain subcritical following a LOCA.

   The impacts of TPBARs on the radiological consequences for all evaluated events are very small, and they remain within 10 CFR 100 regulatory limits. The additional offsite doses due to tritium are small with respect to LOCA source terms and are well within regulatory limits.

   The TPBAR could result in an increase in combustible gas released to the containment in a large break LOCA. This increase was found to be approximately 1474 scf which remains within the capability of the recombiners.

   Analysis has shown that TPBARs are not expected to fail during Condition I through IV events. TPBARs may fail during a LBLOCA or as a result of fuel handling accident. The radiological consequences of these events are within 10 CFR 100 limits. Therefore, there is no significant increase in the consequences of these previously evaluated accidents.

5. TS 4.3.3 - Design Features/Fuel Storage/Capacity

   The Region 2 burnup credit racks described in this TS section are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Due to the deletion of the Region 2 racks, the additional detail provided clarifies existing storage restrictions. Therefore, since they are not installed, there is no increase in the probability or consequences of an accident previously evaluated.

6. TS Bases 3.5.2 - Emergency Core Cooling Systems/ECCS Operating

   Due to the increase of the boron concentration in the RWST and the accumulators, initial mixed boron concentrations are higher and the precipitation concentration is reached sooner. As a result, the hot leg switchover is being shortened. However, the time being shortened does not change the switchover function. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

7. TS Bases 3.6.7 - Hydrogen Recombiners

   This change is administrative in nature and involves only identifying another source of hydrogen gas (tritium) to the bases. The functions for the hydrogen recombiners remain the same. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

8. TPBAR Consolidation Activity

   TPBAR consolidation and associated handling activities are designed to be consistent with the existing fuel handling and heavy load handling processes and equipment currently utilized at the facility, and are designed to preclude increased probability of an accident previously evaluated.

   Consequences of a fuel handling accident for fuel containing TPBARs is evaluated and does not result in exceeding 10 CFR Part 100 limits for off-site dose. All consolidation and heavy load handling activities are designed such that the current fuel handling accident scenario remains bounding. Therefore the consequences of an accident previously evaluated remains within acceptable limits.

B. **<u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>**

1. TS 3.5.1 - Cold Leg Accumulator - Boron Concentration Increase

   The change to the accumulator concentration does not cause the initiation of any accident nor create any new credible limiting single failure. The change does not result in a condition where the design, material, and construction standards of the accumulators and other potentially affected NSSS components, that were applicable prior to the changes, are altered. The integrity and operability of the stainless steel surfaces in the accumulator and affected NSSS components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained during a LOCA since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials.

   The changes in the concentrations increase the amount of boron in the sump following a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs. Thus, there will be no boron precipitation in the core following a LOCA.

   All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed change has no adverse affect on any safety-related system or component and does not challenge the performance or integrity of any safety related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

2. TS 3.5.4 and associated TS Bases Page - RWST - Boron Concentration Increase

The change to the RWST concentration does not cause the initiation of any accident nor create any new credible limiting single failure. The change does not result in a condition where the design, material, and construction standards of the RWST and other potentially affected NSSS components, that were applicable prior to the changes, are altered. The integrity and operability of the stainless steel surfaces in the RWST and affected NSSS components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained during a LOCA since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials.

The changes in the concentrations increase the amount of boron in the sump following a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs. Thus, there will be no boron precipitation in the core following a LOCA.

All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed change has no adverse affect on any safety-related system or component and does not challenge the performance or integrity of any safety related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. TS 3.7.15 and associated TS Bases Pages - Plant Systems/Spent Fuel Assembly Storage

The Region 2 burnup credit racks described in section 4.3.3 are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Therefore, since they are not installed, this change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

4. TS 4.2.1 - Design Features/Reactor Core/Fuel Assemblies

TPBARS have been designed to be compatible with existing Westinghouse 17x17 fuel assemblies and conventional Burnable Poison Rod Assembly (BPRA) handling tools, equipment, and procedures, and therefore, no new accidents or equipment malfunctions are created by the handling of TPBARs. Consolidation activities are discussed separately in Enclosure 5.

TPBARs use materials with known and predictable performance characteristics and are compatible with pressurized water reactor (PWR) coolant. The TPBAR design has specifically included material similar to those used in standard burnable absorber rods with the exception of internal assemblies used in the production and retention of tritium. As described in the TPC Topical Report, these materials are compatible with the reactor coolant system (RCS) and core design. Therefore, no new accidents or equipment malfunctions are created by the presence of the TPBARs in the RCS.

Mechanical design criteria have been established to ensure that TPBARs will not fail during Condition I or II events. Analysis has shown that TPBARs, appropriately positioned in the core operate within the established thermal-hydraulic criteria. Due to the expected high reliability of TPBAR components the frequency of TPBAR cladding failures is very small, such that multiple adjacent TPBAR failures in limiting locations is not considered credible. In addition, analysis has shown that if a single TPBAR fails catastrophically in a high power location during normal operation and the lithium is leached out, the global reactivity increase is negligible and the local power peaking is small enough that DNBR limits and fuel rod integrity are not challenged. Therefore, no new accidents or equipment malfunctions are created by the presence of the TPBARs in the reactor.

Analysis has shown that TPBARs will not fail during Condition III and IV events. TPBARs may fail during a cold leg large break loss-of-coolant-accident or as a result of a fuel handling accident. The radiological consequences of these events are within 10 CFR 100 limits. Therefore, there is no significant increase in consequences of these previously evaluated accidents.

TPBARs do not adversely affect reactor neutronic or thermal-hydraulic performance; therefore they do not create the possibility of accidents or equipment malfunctions of a different type than previously evaluated while in the reactor.

5. TS 4.3.3 - Design Features/Fuel Storage/Capacity

   The Region 2 burnup credit racks described in this section are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Due to the deletion of the Region 2 racks, the additional detail provided clarifies existing storage restrictions. Therefore, since they are not installed, this change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

6. Bases 3.5.2 - Emergency Core Cooling Systems/ECCS Operating

   Due to the increase of the boron concentration in the RWST and the accumulators, initial mixed boron concentrations are higher and the precipitation concentration is reached sooner. As a result, the hot leg switchover value is being shortened. This time being shortened does not change the switchover function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

7. Bases 3.6.7 - Hydrogen Recombiners

   This change is administrative in nature and only involves only identifying another source of hydrogen gas (tritium) to the bases. The functions for the hydrogen recombiners remain the same. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

8. TPBAR Consolidation Activity -

   The consolidation and handling activities are bounded by current fuel handling evaluations. Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

C. **<u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>**

1. TS 3.5.1 - Cold Leg Accumulator - Boron Concentration Increase

   The change does not invalidate any of the non-LOCA safety analysis results or conclusions, and all of the non-LOCA safety analysis acceptance criteria continue to be met. The licensing basis SBLOCA analyses does not credit the accumulator boron and is not affected by the proposed change. Therefore, there is no reduction in the margin to the Peak clad temperature (PCT) limit for the SBLOCA. There is no increase in the Best Estimate LBLOCA PCT; therefore, there continues to be a high level of probability that the ECCS acceptance criteria limit is not exceeded with regard to the LBLOCA analysis. The increased boron concentration is sufficient to maintain subcriticality during the LBLOCA, and a post-LOCA long term core cooling analysis demonstrated that the post-LOCA sump boron concentration is sufficient to prevent recriticality. The revised hot leg switchover time, which will be implemented in the EOPs, will prevent boron precipitation. The licensing basis containment and SLB M&E releases remain bounding, and the SGTR event acceptance criteria continue to be met. Furthermore, the changes do not affect the safety related performance of the accumulator or related NSSS components. Therefore, there is no significant reduction in the margin of safety.

2. TS 3.5.4 and associated TS Bases Page- RWST - Boron Concentration Increase

   The change does not invalidate any of the non-LOCA safety analysis results or conclusions, and all of the non-LOCA safety analysis acceptance criteria continue to be met. The licensing basis SBLOCA analyses does not credit the RWST boron and is not affected by the proposed change. Therefore, there is no reduction in the margin to the PCT limit for the SBLOCA. There is no increase in the Best Estimate LBLOCA PCT; therefore, there continues to be a high level of probability that the ECCS acceptance criteria limit is not exceeded with regard to the LBLOCA analysis. The increased boron concentration is sufficient to prevent recriticality. The revised hot leg switchover time, which will be implemented in the EOPs, will prevent boron precipitation. The licensing basis containment and SLB M&E releases remain bounding, and the SGTR event acceptance criteria continue to be met. Furthermore, the changes do not affect the safety related performance of the RWST or related NSSS components. Therefore, there is no significant reduction in the margin of safety.

3. TS 3.7.15 and associated TS Bases Pages - Plant Systems/Spent Fuel Assembly Storage

   The Region 2 burnup credit racks described in section 4.3.3 are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Therefore, since they are not installed, this change would not involve a reduction in a margin of safety.

4. TS 4.2.1 - Design Features/Reactor Core/Fuel Assemblies

   TPBARs have been designed to be compatible with existing fuel assemblies. TPBARs do not adversely affect reactor neutronic or thermal-hydraulic performance. Analysis indicates that reactor core behavior and offsite doses remain relatively unchanged. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

5. TS 4.3.3 - Design Features/Fuel Storage/Capacity

   The Region 2 burnup credit racks described in section 4.3.3 are not currently installed in the plant. Since the time that these racks were licensed, TVA has determined not to install or utilize this storage option. Due to the deletion of the Region 2 racks, the additional detail provided clarifies existing storage restrictions and does not reduce the margin of safety in existing storage requirements. Therefore, since they are not installed, this change would not involve a reduction in a margin of safety.

6. Bases 3.5.2 - Emergency Core Cooling Systems/ECCS Operating

   Due to the increase of the boron concentration in the RWST and the accumulators, initial mixed boron concentrations are higher and the precipitation concentration is reached sooner. As a result, the hot leg switchover value is being shortened. This time being shortened does not change the switchover function. Therefore, this change does not involve a reduction in the margin of safety.

7. Bases 3.6.7 - Hydrogen Recombiners

   This change is administrative in nature and only involves only identifying another source of hydrogen gas (tritium) in the bases. The functions for the hydrogen recombiners remain the same. Therefore, this change does not involve a reduction in the margin of safety.

8. TPBAR Consolidation Activity

   The changes do not significantly affect the safety related performance of any plant operations, system, structures, or components. The consolidation activity is bounded by current fuel handling evaluations. Therefore, there is no significant reduction in the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The environmental impacts of producing tritium in TVA's Watts Bar Unit 1 were assessed in a 1999 "Final Environmental Impact Statement (EIS) for the Production of Tritium in a Commercial Light Water Reactors" (DOE/EIS-0288) prepared by Department of Energy (DOE). TVA was a cooperating agency in the preparation of this EIS. In accordance with 40 CFR 1506.3(c) of the Council on Environmental Quality regulations, TVA independently reviewed the EIS prepared by DOE, found it to be adequate, and adopted the EIS. TVA's "Record of Decision and Adoption of the Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor" was published in the Federal Register at 65 Federal Register 26259 (May 5, 2000). As part of the process of developing this Tritium Program license amendment request, TVA conducted a contemporaneous review of the DOE EIS and TVA's Record of Decision, focusing on any changes in radiological impacts associated with the program. That review determined that there were no substantial changes in the Tritium Program since the publication of the 1999 EIS that were relevant to new circumstances or information relevant to environmental concerns which were bearing on the Tritium program or its impacts.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

TECHNICAL SPECIFICATIONS (TS)
PAGE MARKUPS

<u>MARKED-UP TECHNICAL SPECIFICATION PAGES</u>

3.5-2
3.5-10
3.7-31
3.7-32
4.0-1
4.0-2
4.0-3
4.0-4
4.0-7
4.0-9
4.0-10

B 3.5-10
B 3.5-26
B 3.6-44
B 3.7-75
B 3.7-76
B 3.7-77

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7630 gallons and ≤ 8000 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ $\boxed{2400}$ ppm and ≤ $\boxed{2700}$ ppm. 3800 3500	31 days __AND__ -----NOTE------ Only required to be performed for affected accumulators -------------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	----------------NOTE-------------------- Only required to be performed when ambient air temperature is < 60°F or > 105°F. -- Verify RWST borated water temperature is ≥ 60°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ $\boxed{2500}$ ppm and ≤ $\boxed{2700}$ ppm.	7 days

3600 3800

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 1 or Region 2 shall be within the Acceptable Burnup Domain of Figure 3.7.15-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Initiate action to move the noncomplying fuel assembly.	Immediately

DELETE

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure 3.7.15-1 or~~ Specification 4.3.1.1.	Prior to storing the fuel assembly.

REPLACE WITH

The combination of initial enrichment and burnup of each spent fuel assembly stored shall be in accordance with Specification 4.3.1.1.

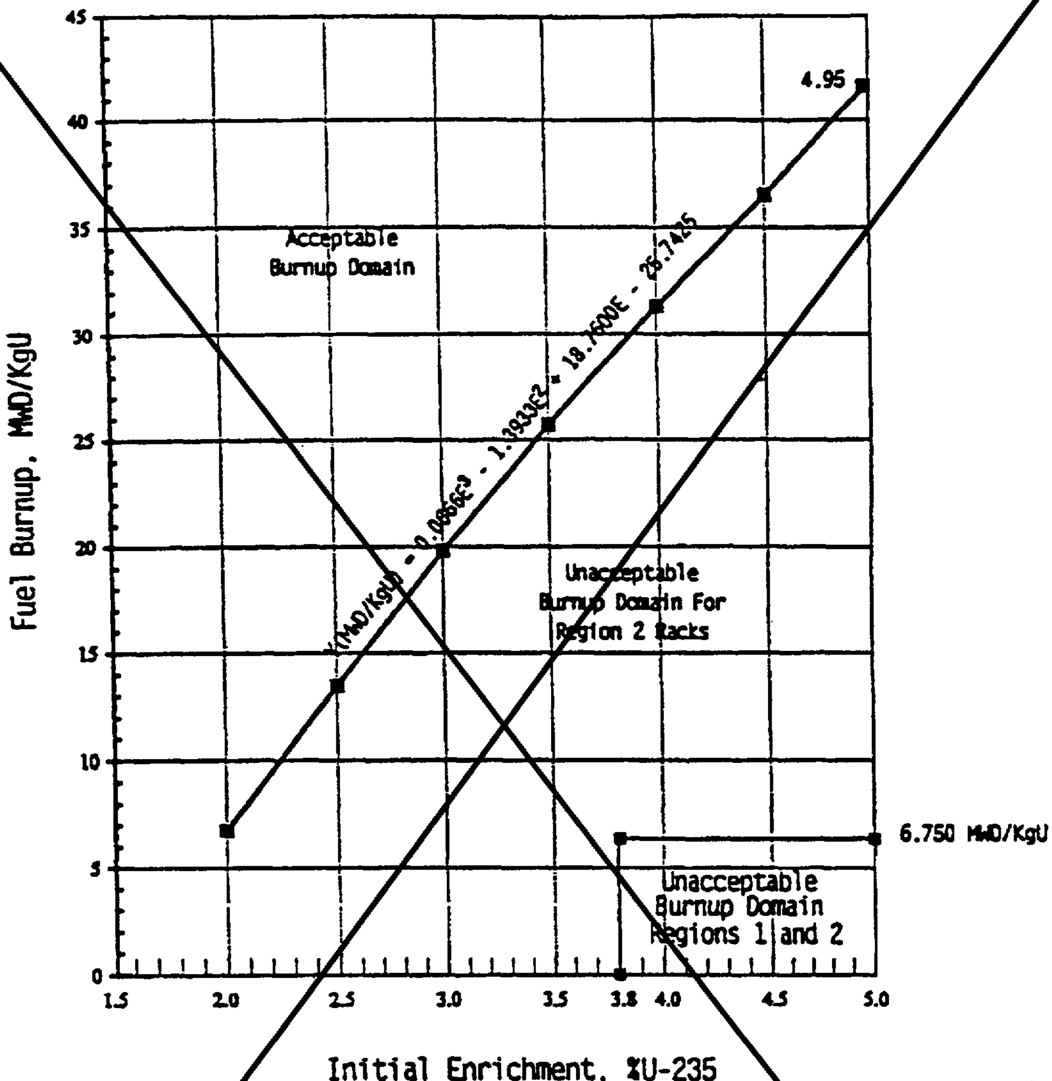

Figure 3.7.15-1 Acceptable Burnup Domain - Watts Bar
Spent Fuel Storage Racks

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4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or Zirlo fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. For Unit 1, Cycle 2, Watts Bar is authorized to place a limited number of Tritium Producing Burnable Absorber Rod lead test assemblies into the reactor in accordance with TVA's application dated April 30, as supplemented June 18, July 21 (3 letters), and August 7 and 21, 1997.

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

REPLACE WITH:

For Unit 1, Watts Bar is authorized to place a maximum of 2304 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

INSERT
are

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

DELETE

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR;

c. Distances between fuel assemblies ~~as follows:~~

~~1. A nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules (Region 1).~~

DELETE

~~2. A nominal 8.972 inch center-to-center spacing in the ten burnup credit rack modules peripherally located adjacent to the south and west pool walls (Region 2); and~~

~~d. Spent fuel assemblies with a burnup in the "acceptable burnup domain" of Figure 3.7.15-1 may be allowed unrestricted storage in either type of fuel storage rack.~~

~~e. New or partially spent fuel assemblies with a burnup in the "unacceptable burnup domain" of Figure 3.7.15-1 will be stored in compliance with the following configuration:~~

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

DELETE

INSERT

d. Fuel

DELETE

INSERT A
SEE NEXT PAGE.

~~1. In the flux trap rack modules (Region 1)~~ fuel assemblies with enrichments less than or equal to 3.80 weight percent U-235 are allowed unrestricted storage. ~~Fuel assemblies with enrichment greater than 3.80 weight percent U-235 and burnup less than 6.750 megawattday/kilogram uranium (MWD/KgU) shall be placed in storage cells that face adjacent cells in the flux trap modules containing either water or fuel assemblies with accumulated burnup of at least 20 MWD/KgU.~~

~~2. Storage in any burnup credit rack modules (Region 2) located in the pool as well as in the fuel cask loading area is restricted to fuel of 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity in the fuel racks. The minimum required assembly average burnup is given by Y(MWD/KgU) where Y = 0.0666E³ - 1.8933E² + 18.7600E - 25.7425, where E is the initial enrichment in the axial zone of highest enrichment. Figure 3.7.15-1 illustrates the burnup enrichment equation in graphical form.~~

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

(continued)

INSERT A

e. Fuel assemblies with initial enrichments greater than 3.80 weight percent and less than a maximum of 5 percent enrichment (nominally 4.95± 0.05 percent) may be stored in the spent fuel racks in one of four arrangements with specific limits as identified below:

1. Spent fuel assemblies may be stored in the racks without further restrictions provided the burnup of each assembly is in the acceptable domain identified in Figure 4.3-3, depending upon the specified initial enrichment.

2. New and spent fuel assemblies may be stored in a checkerboard arrangement of 2 new and 2 spent assemblies, provided that each spent fuel assembly has accumulated a minimum burnup in the acceptable domain identified in Figure 4.3-4.

3. New fuel assemblies may be stored in 4-cell arrays with 1 of the 4 cells remaining empty of fuel (i.e. containing only water or water with up to 75 percent by volume of non-fuel bearing material).

4. New fuel assemblies with a minimum of 32 integral fuel burnable absorber (IFBA) rods may be stored face adjacent without further restriction, provided the loading of ZrB_2 in the coating of each IFBA rod is minimum of 1.25x. (1.9625 mg/in.)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

a. Fuel assemblies having a maximum enrichment of 5.0 weight percent U-235 and shall be maintained with the arrangement of 120 storage locations shown in Figure 4.3-2:

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR:

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR: and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Elevation 747 feet - 1 1/2 inches.

DELETE

4.3.3 Capacity

~~The total spent fuel storage capacity is 1610 fuel assemblies.~~

4.3.3.1 The primary portion of the spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in 24 flux trap rack modules.

4.3.3.2 No more than 224 fuel assemblies will be stored in ten smaller burnup credit rack modules peripherally located adjacent to the south and west walls of the pool.

(continued)

REPLACE WITH NEXT PAGE

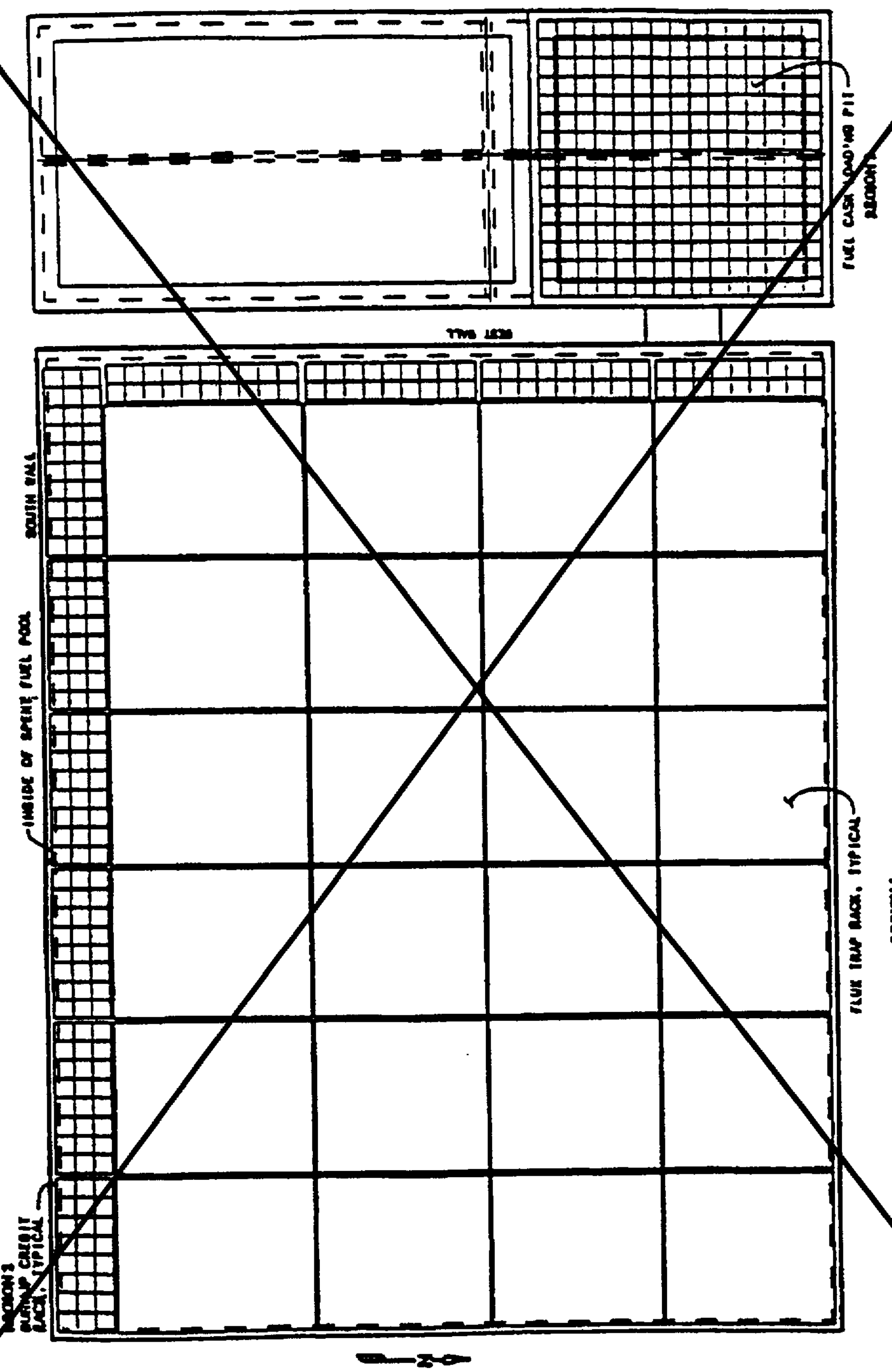
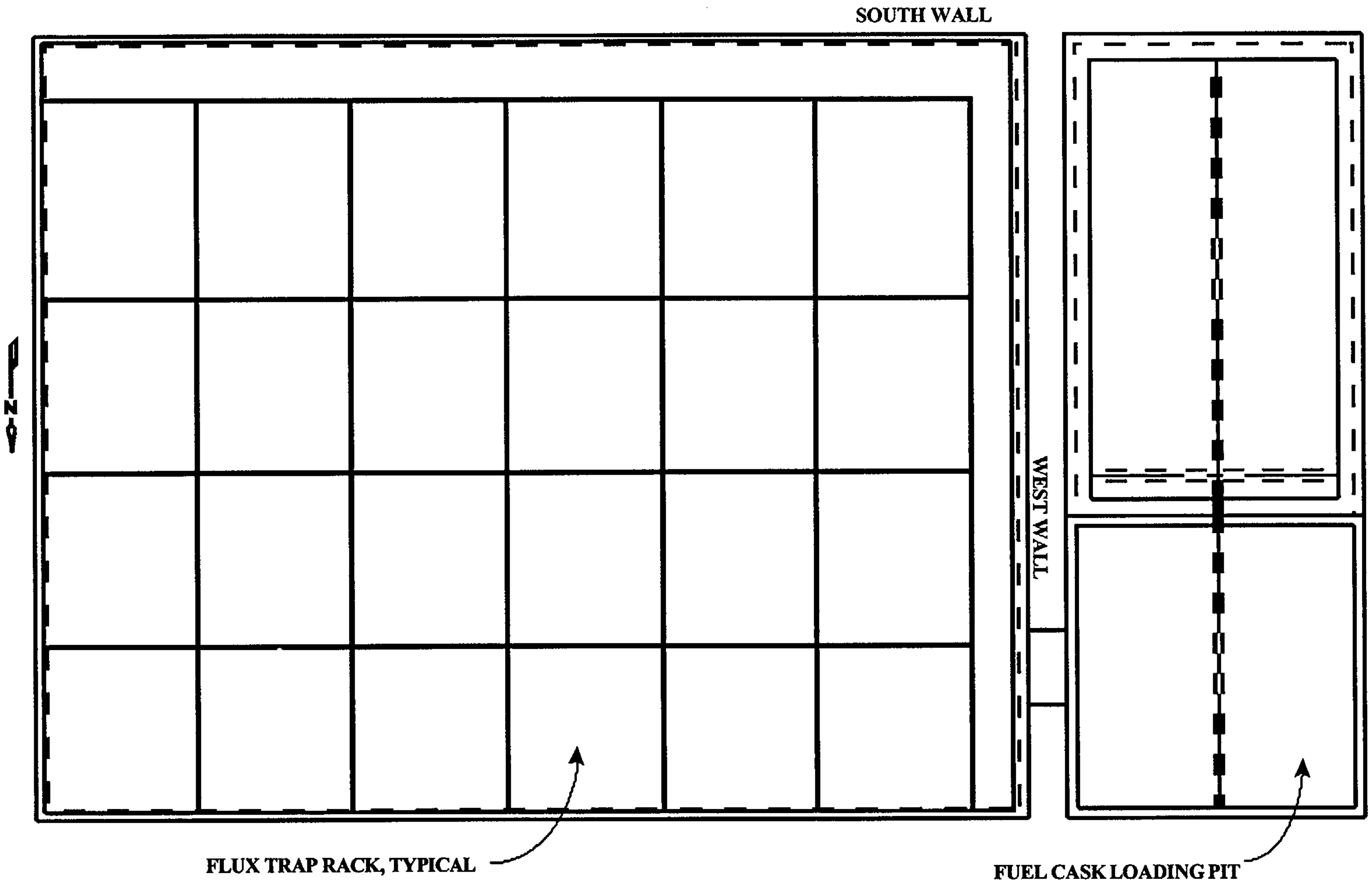


FIGURE 4.3-1
SPENT FUEL STORAGE RACKS



PLAN
SPENT FUEL POOL

FIGURE 4.3-1
SPENT FUEL STORAGE RACKS

INSERT NEW FIGURE

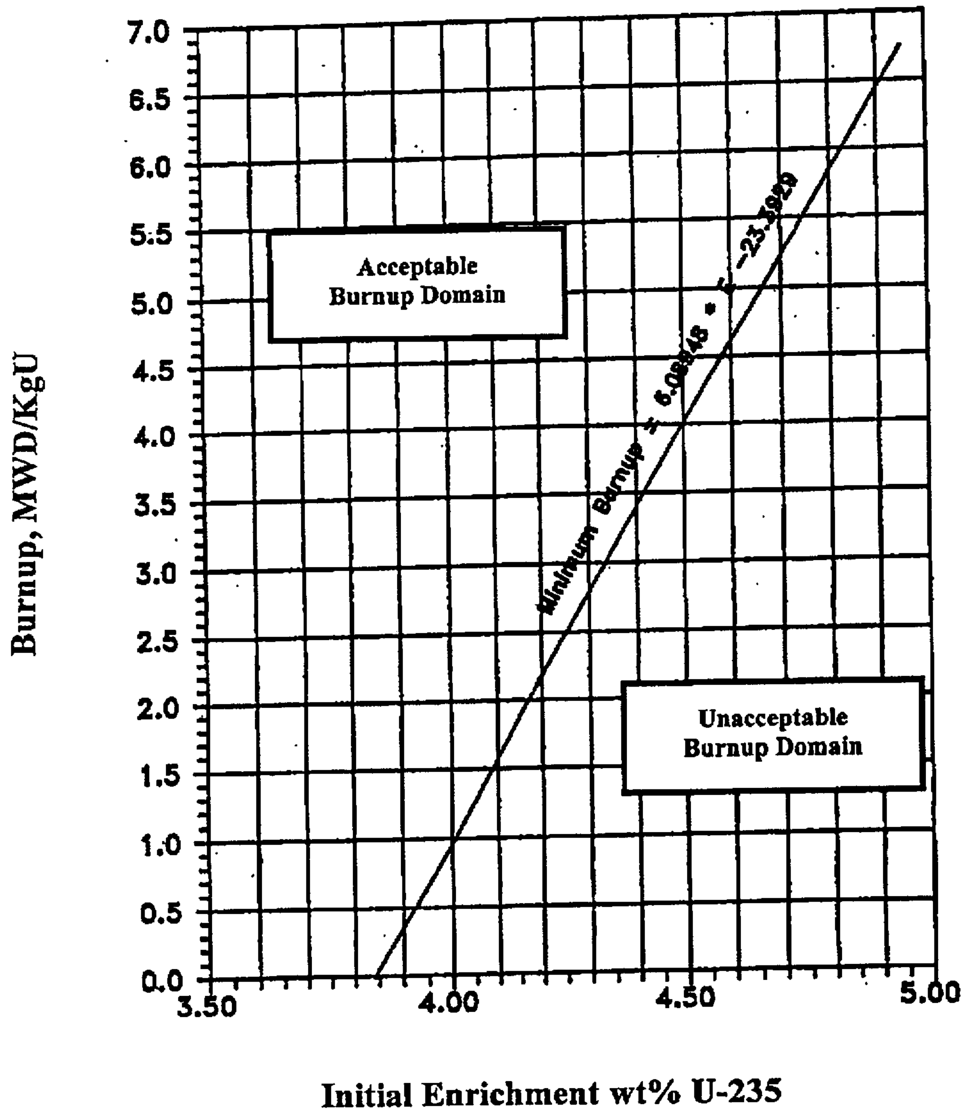


Figure 4.3-3
Minimum Required Burnup for Unrestricted Storage of Spent Fuel of Various Initial Enrichments

INSERT NEW FIGURE

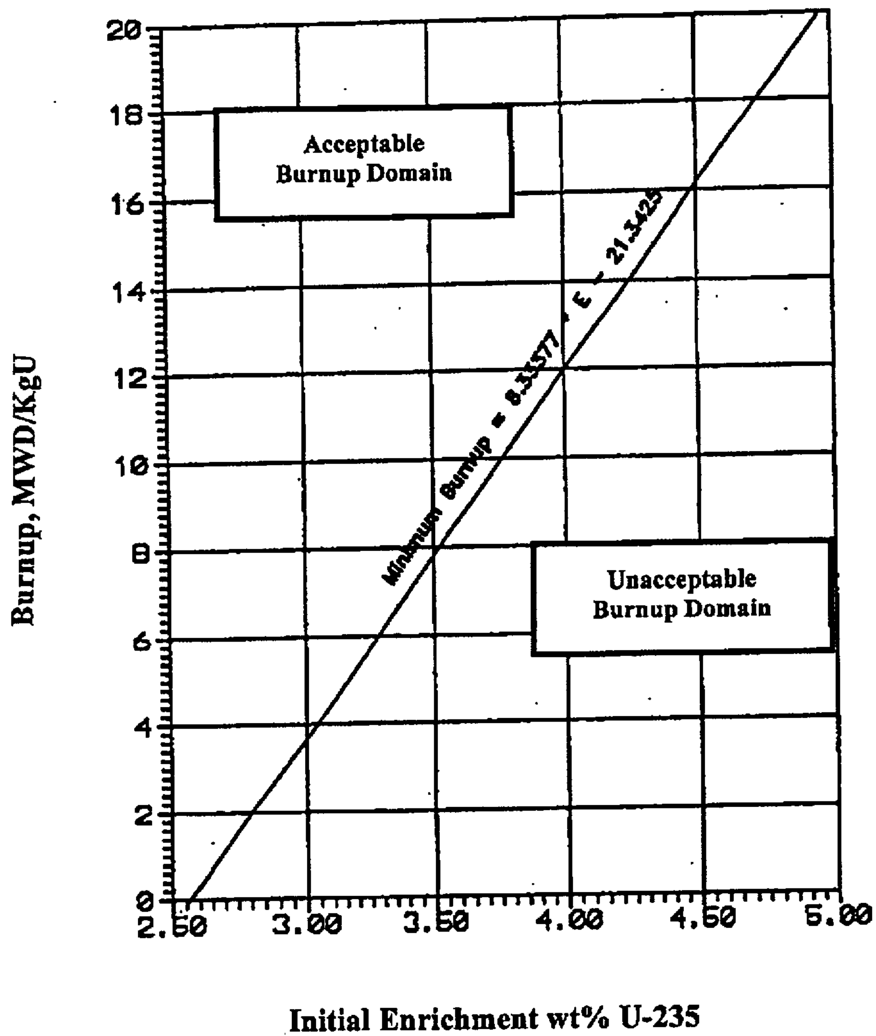


Figure 4.3-4
Minimum Required Burnup for 2x2 Checkerboard Arrangement of 2 Spent Fuel Assemblies with 2 New Fuel Assemblies of 5% Enrichment (Maximum)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

5.5

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 9 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

3600

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of 2500 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2700 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

3800

(continued)

BASES

REPLACE WITH:

A metal steam reaction between the zirconium fuel rod cladding, TPBAR zirconium internals, and the reactor coolant;

LE NALYSES nued)

Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

REPLACE WITH:

Hydrogen in the RCS at the time of LOCA (i.e., hydrogen dissolved in the reactor coolant, hydrogen gas in pressurizer vapor space, and tritium contained in TPBARS); or

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.4 v/o about 5 days after the LOCA and 4.1 v/o about 2 days later if no recombiner was functioning (Ref. 5). Initiating the hydrogen recombiners within 24 hours after a DBA will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4).

REPLACE WITH

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 4.0 v/o in about 3 days if no recombiner was functioning (Ref. 5). Initiating the hydrogen recombiners within 24 hours after a DBA will maintain the hydrogen concentration in the primary containment below flammability limits.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

INSERT The spent fuel pool contains

INSERT rack

DELETE

~~In the spent fuel storage design (References 1 and 2), the spent fuel pool area is divided into two separate and distinct regions for criticality considerations.~~ Region 1, flux trap modules with 1386 storage positions, is designed to accommodate fuel with enrichment as high as 3.8 weight percent U-235 without restrictions. Storage of fuel assemblies with enrichment between 3.8 and 5.0 weight percent ~~in Region 1~~ requires either fuel burnup ~~of ≥26,750 MWD/KgU~~ in accordance with Figure 3.7.15-1, or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU in accordance with paragraph 4.3.1.1.

REPLACE WITH and are

DELETE

REPLACE WITH specification 4.3.1.1

DELETE

~~Region 2 burnup credit rack modules, with 224 total storage positions, is designed to accommodate fuel with 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU or, in accordance with Figure 3.7.15-1, fuel of lower enrichment which yields an equivalent reactivity~~

REPLACE WITH racks

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design ~~of both regions~~ is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975, and the April 1978 NRC letter (Reference 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, an

REPLACE WITH 1

REPLACE WITH the storage racks

BASES

BACKGROUND (continued)

abnormal scenario could be associated with the improper movement of a relatively high enrichment, low exposure fuel assembly from Region 1 to Region 2, or the misloading of a fuel assembly in either region. This ~~could potentially~~ increase the criticality of the storage regions. To mitigate these postulated criticality-related events, boron is dissolved in the pool water. Safe operation of the ~~spent~~ fuel storage design with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly in the pool, it is necessary to perform SR 3.9.9.1.

DELETE

INSERT

loading

REPLACE WITH

racks

APPLICABLE SAFETY ANALYSES

The hypothetical events can only take place during or as a result of the movement of an assembly. For these occurrences, the presence of soluble boron in the spent fuel storage pool, (controlled by LCO 3.9.9, "Spent Fuel Pool Boron Concentration,") prevents criticality in both storage rack regions. By closely controlling the movement of each ~~assembly and by~~ checking the location of each assembly after movement, the time period for potential occurrences may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

REPLACE WITH

racks.

REPLACE WITH

the

REPLACE WITH

specification 4.3.1.1

DELETE

LCO

The restrictions on the placement of ~~fuel assemblies within~~ the spent fuel pool in accordance with Figure 3.7.15-1, in the accompanying LCO, ensures the k_{eff} ~~will always remain~~ ≤ 0.95, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

BASES (continued)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If unable to move irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in Mode 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

DELETE

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with ~~Figure 3.7.15-1~~ or paragraph 4.3.1.1 the immediate action is to initiate action to make the necessary fuel assembly movements to bring the configuration into compliance with ~~Figure 3.7.15-1 or~~ Specification 4.3.1.1.

INSERT Specification 4.3.1.1

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 in the accompanying LCO. ~~For fuel assemblies in the unacceptable range of Figure 3.7.15-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

REFERENCES

1. Watts Bar FSAR, Sections 4.3.2.7 and 9.1.2.

2. Spent Fuel Pool Modification for Increased Storage Capacity, (Chapter 4), Watts Bar Unit 1, submitted by TVA letter dated October 23, 1996.

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

DELETE

REPLACE WITH 1

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

TECHNICAL SPECIFICATIONS (TS)
REVISED PAGES

<u>REVISED/ADDED TECHNICAL SPECIFICATION PAGES</u>

3.5-2
3.5-10
3.7-31
3.7-32
4.0-1
4.0-2
4.0-3
4.0-4
4.0-7
4.0-9 - ADDED
4.0-10 - ADDED

B 3.5-10
B 3.5-26
B 3.6-44
B 3.7-75
B 3.7-76
B 3.7-77

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7630 gallons and ≤ 8000 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 3500 ppm and ≤ 3800 ppm.	31 days AND -----NOTE------ Only required to be performed for affected accumulators --------------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	----------------NOTE-------------------- Only required to be performed when ambient air temperature is < 60°F or > 105°F. -------------------------------------- Verify RWST borated water temperature is ≥ 60°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 3600 ppm and ≤ 3800 ppm.	7 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored shall be in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Initiate action to move the noncomplying fuel assembly.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Specification 4.3.1.1.	Prior to storing the fuel assembly.

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4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or Zirlo fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. For Unit 1, Watts Bar is authorized to place a maximum of 2304 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR;

c. Distances between fuel assemblies are a nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules.

d. Fuel assemblies with enrichments less than or equal to 3.80 weight percent U-235 are allowed unrestricted storage.

e. Fuel assemblies with initial enrichments greater than 3.80 weight percent and less than a maximum of 5 percent enrichment (nominally 4.95± 0.05 percent) may be stored in the spent fuel racks in one of four arrangements with specific limits as identified below:

1. Spent fuel assemblies may be stored in the racks without further restrictions provided the burnup of each assembly is in the acceptable domain identified in Figure 4.3-3, depending upon the specified initial enrichment.

2. New and spent fuel assemblies may be stored in a checkerboard arrangement of 2 new and 2 spent assemblies, provided that each spent fuel assembly has accumulated a minimum burnup in the acceptable domain identified in Figure 4.3-4.

3. New fuel assemblies may be stored in 4-cell arrays with 1 of the 4 cells remaining empty of fuel (i.e. containing only water or water with up to 75 percent by volume of non-fuel bearing material.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4. New fuel assemblies with a minimum of 32 integral fuel burnable absorber (IFBA) rods may be stored face adjacent without further restriction, provided the loading of ZrB_2 in the coating of each IFBA rod is minimum of 1.25x (1.9625 mg/in).

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements. A water cell is defined as a cell containing water or non-fissile material with no more than 75% of the water displaced.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum enrichment of 5.0 weight percent U-235 and shall be maintained with the arrangement of 120 storage locations shown in Figure 4.3-2;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

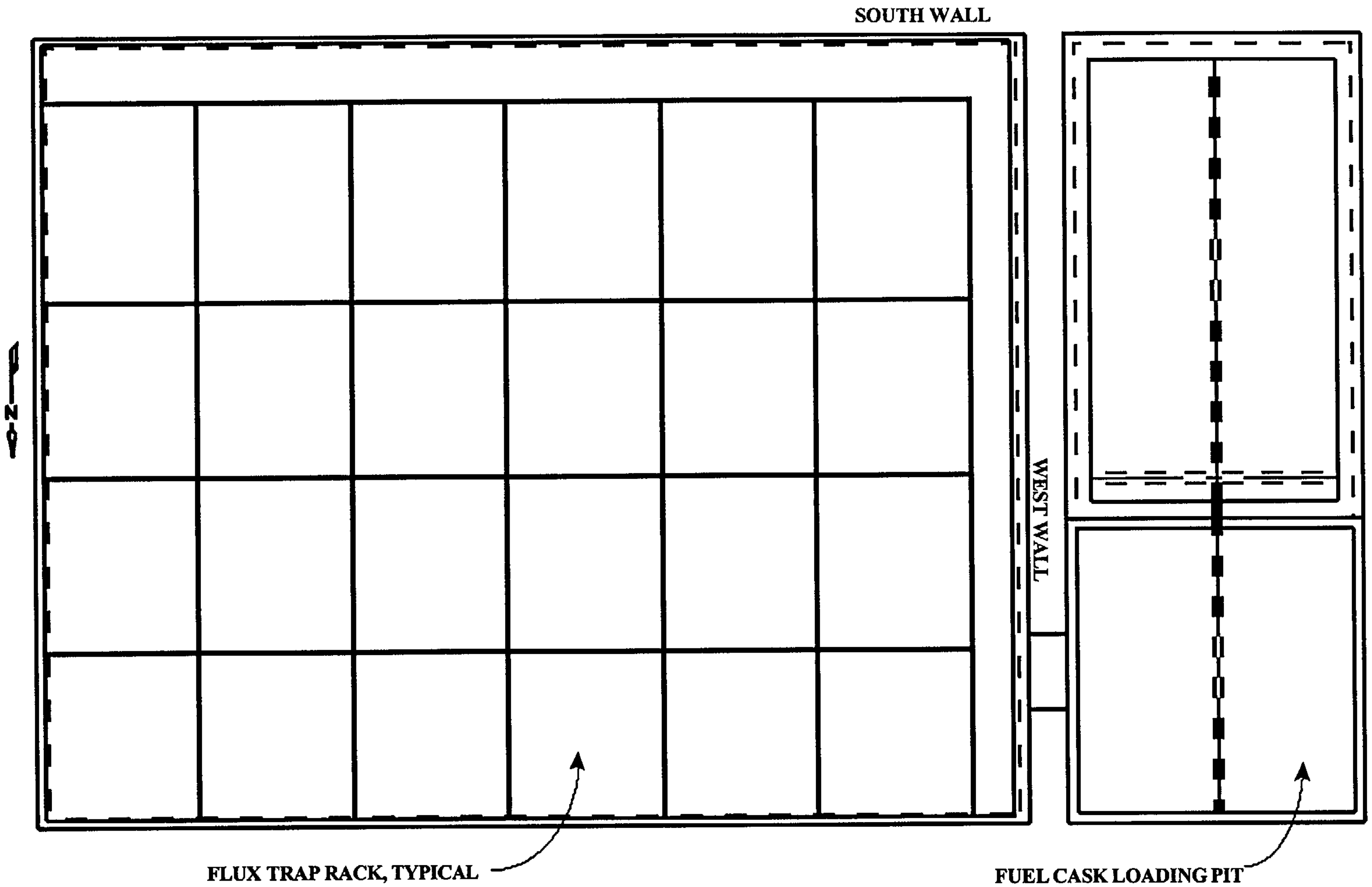
4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Elevation 747 feet - 1 1/2 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in 24 flux trap rack modules.

(continued)



PLAN
SPENT FUEL POOL

FIGURE 4.3-1
SPENT FUEL STORAGE RACKS

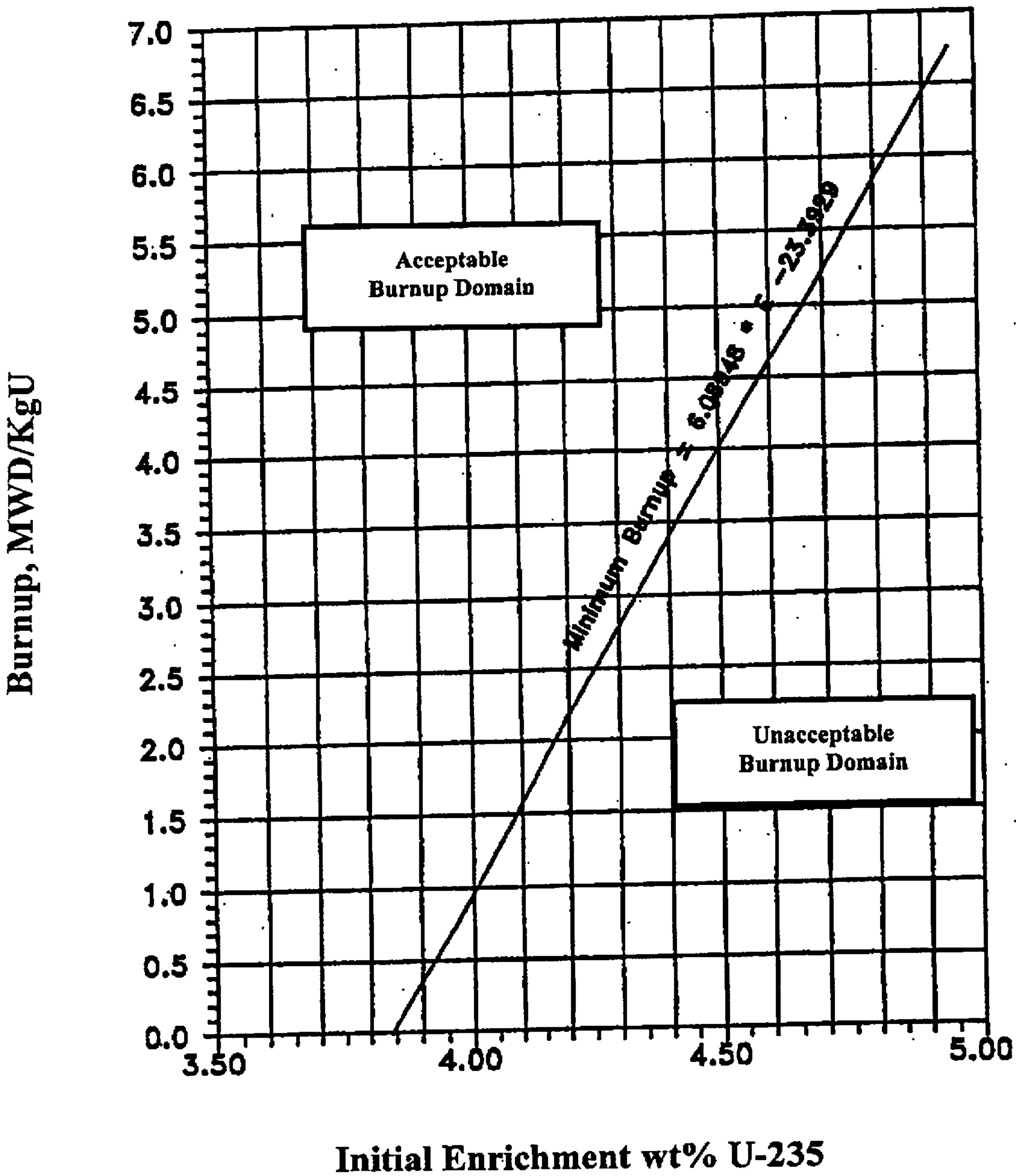


Figure 4.3-3
Minimum Required Burnup for Unrestricted Storage of Spent Fuel of Various Initial Enrichments

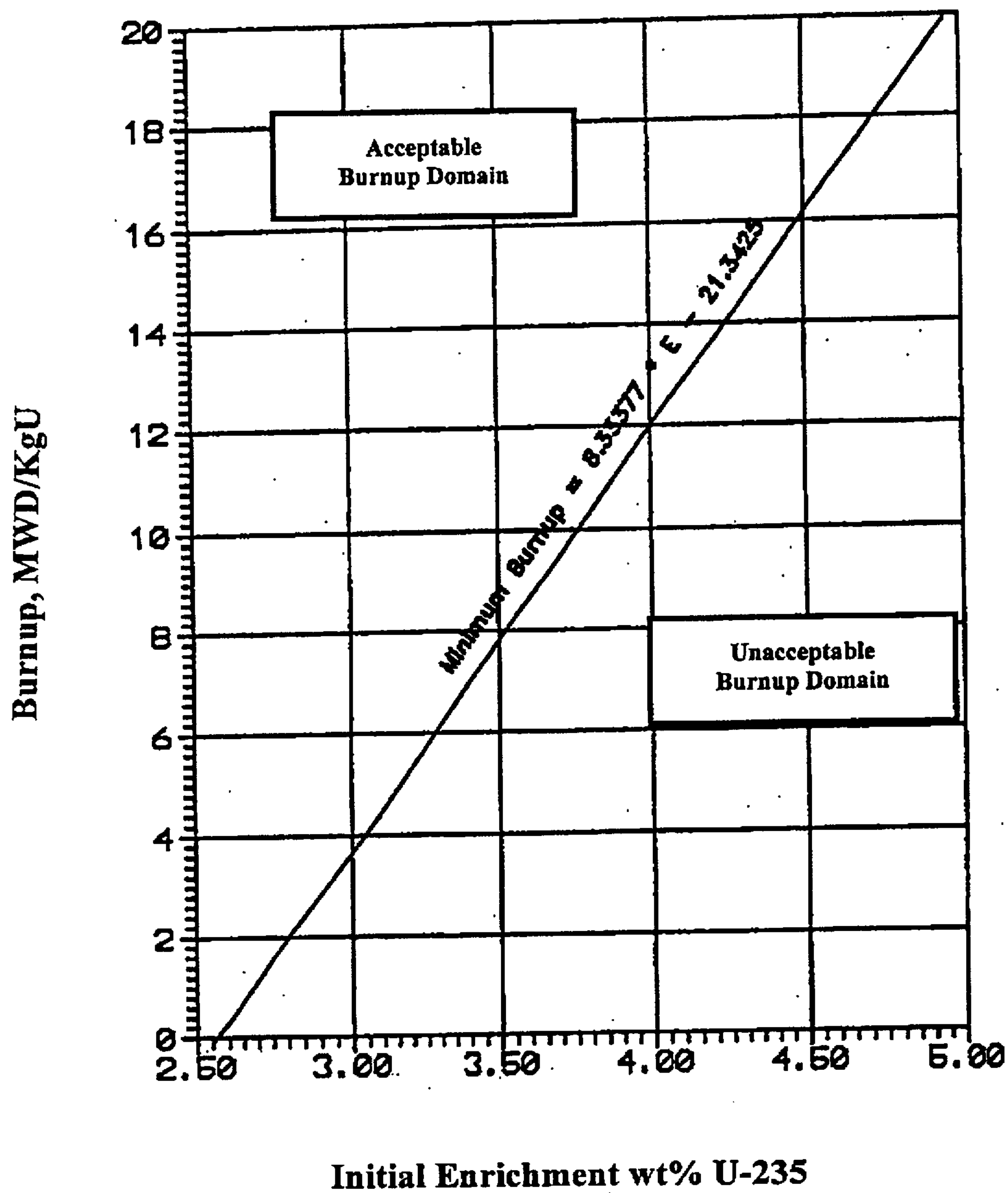


Figure 4.3-4
Minimum Required Burnup for 2x2 Checkerboard Arrangement of 2 Spent Fuel Assemblies with 2 New Fuel Assemblies of 5% Enrichment (Maximum)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 5.5 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of 3600 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 3800 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding, TPBAR zirconium internals, and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of LOCA (i.e., hydrogen dissolved in the reactor coolant, hydrogen gas in pressurizer vapor space, and tritium contained in TPBARs); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 4.0 v/o in about 3 days if no recombiner was functioning (Ref. 5). Initiating the hydrogen recombiners within 24 hours after a DBA will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4).

The hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel pool contains flux trap rack modules with 1386 storage positions and are designed to accommodate fuel with enrichment as high as 3.8 weight percent U-235 without restrictions. Storage of fuel assemblies with enrichment between 3.8 and 5.0 weight percent requires either fuel burnup in accordance with Specification 4.3.1.1 or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU in accordance with Specification 4.3.1.1.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains the storage racks in a subcritical condition during normal operation with the racks fully loaded. The double contingency principle discussed in ANSI N-16.1-1975, and the April 1978 NRC letter (Reference 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, an

(continued)

BASES

BACKGROUND (continued)

abnormal scenario could be associated with the improper loading of a relatively high enrichment, low exposure fuel assembly. This could potentially increase the criticality of the storage racks. To mitigate these postulated criticality-related events, boron is dissolved in the pool water. Safe operation of the spent fuel storage design with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly in the pool, it is necessary to perform SR 3.9.9.1.

APPLICABLE SAFETY ANALYSES

The hypothetical events can only take place during or as a result of the movement of an assembly. For these occurrences, the presence of soluble boron in the spent fuel storage pool, (controlled by LCO 3.9.9, "Spent Fuel Pool Boron Concentration,") prevents criticality in the storage racks. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential occurrences may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

The restrictions on the placement of fuel assemblies within the spent fuel pool in accordance with Specification 4.3.1.1 in the accompanying LCO, ensures the k_{eff} will always remain ≤ 0.95, assuming the pool to be flooded with unborated water.

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

(continued)

BASES (continued)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If unable to move irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in Mode 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with Specification 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movements to bring the configuration into compliance with Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Specification 4.3.1.1 in the accompanying LCO.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1
DOCKET NO. 390

WESTINGHOUSE REPORT NDP-00-0344

This enclosure includes errata pages for WESTINGHOUSE REPORT NDP-00-0344, Pages 1-26 and 2-59.

NDP-00-0344, Revision 1

IMPLEMENTATION AND UTILIZATION OF

TRITIUM PRODUCING BURNABLE ABSORBER RODS (TPBARS)

IN WATTS BAR UNIT 1

July 2001

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LIST OF ACRONYMS

AFD	Axial Flux Difference
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
ARI	All Rods In
ARO	All Rods Out
ART	Adjusted Reference Temperature
ASL	Acceptable Suppliers List
ATWS	Anticipated Transients Without Scram
BA	Burnable Absorber
BOL	Beginning of Life
BP	Burnable Poison
BPRA	Burnable Poison Rod Assembly (WABA and/or Pyrex Burnable Absorber)
CCS	Component Cooling System
CFR	Code of Federal Regulations
CLWR	Commercial Light Water Reactor
COLR	Core Operating Limits Report
COMS	Cold Overpressure Mitigation System
CRDM	Control Rod Drive Mechanism
CRDS	Control Rod Drive System
CVCS	Chemical and Volume Control System
CZP	Cold Zero Power
DAC	Derived Air Concentration
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
DPC	Doppler-only Power Coefficient
DWMS	Demineralized Water Makeup System
EC	Equilibrium Cycle
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EFPY	Effective Full Power Years
EL	Evaluation Level
EOL	End of Life
EQ	Environmental Qualification
ERCW	Essential Raw Cooling Water
ERG	Emergency Response Guideline
FC	First Cycle
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GVR	Gas Volume Ratio
HFP	Hot Full Power
HTC	Heat Transfer Coefficient
HZP	Hot Zero Power

LIST OF ACRONYMS (cont.)

ID	Inner Diameter
IFBA	Integral Fuel Burnable Absorber
INEEL	Idaho National Engineering and Environmental Laboratory
LAR	License Amendment Request
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LTOPS	Low Temperature, Overpressure Protection System
M/E	Mass and Energy
M&TE	Measuring and Test Equipment
MPH	Material Properties Handbook
MSLB	Main Steamline Break
MTC	Moderator Temperature Coefficient
MWD/MTU	Megawatt Days per Metric Ton of Uranium
MWt	Megawatt Thermal
NDE	Nondestructive Examination
NPZ	Nickel-Plated Zirconium
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OD	Outer Diameter
ODCM	Offsite Dose Calculation Manual
PCM	Percent Milli-rho
PCT	Peak Cladding Temperature
PNNL	Pacific Northwest National Laboratory
PQAP	Package Quality Assurance Plan
P-T	Pressure-Temperature
PTLR	Pressure Temperature Limits Report
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
QA	Quality Assurance
QMS	Quality Management System
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
REP	Radiological Emergency Plan
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RPVSA	Reactor Pressure Vessel System Analysis
RT	Reference Temperature
RV	Reactor Vessel
RWAP	Rod Withdrawal at Power

LIST OF ACRONYMS (cont.)

RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SDM	Shutdown Margin
SER	Safety Evaluation Report
SFP	Spent Fuel Pit or Pool
SFPCS	Spent Fuel Pool Cooling System
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SLB	Steamline Break
SQN	Sequoyah Nuclear Plant
SRP	Standard Review Plan
SS	Stainless Steel
STP	Standard Temperature and Pressure
TCF	TPBAR Consolidation Fixture
TEDE	Total Effective Dose Equivalent
T/H	Thermal/Hydraulic
TPBAR	Tritium Producing Burnable Absorber Rod
TPC	Tritium Production Core
TS	Technical Specification
TTQP	Tritium Target Qualification Project
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
VCT	Volume Control Tank
WABA	Wet Annular Burnable Absorber
WBN	Watts Bar Nuclear Plant
w/o	Weight Percent or Without (depending on context)
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

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

A tritium production core (TPC) topical report (NDP-98-181, Rev. 1) was written that addressed the safety and licensing issues associated with incorporating a full complement of TPBARs in a CLWR, specifically a pressurized water reactor (PWR). The U.S. Nuclear Regulatory Commission's (NRC) Standard Review Plan (SRP) (NUREG-0800) was used as the basis for evaluating the impact of the TPBARs on a reference plant. The NRC reviewed the TPC topical report and issued a Safety Evaluation Report (SER) (NUREG-1672) to support plant specific licensing of TPBARs in a PWR. A number of issues were cited in the TPC topical report and the SER requiring the performance of plant specific evaluations and analyses to demonstrate that no significant safety issues are raised by the operation of a PWR with TPBARs.

This report addresses the required plant specific evaluations and analyses made for WBN to demonstrate that there are no significant safety or operational issues when TPBARs are incorporated into WBN core designs. Specifically, this report:

1. Addresses the 17 plant specific interface issues listed in NUREG-1672, Section 5.1. The following interface items have been submitted previously under a separate cover letter:

   a. Station Service Water

   b. Ultimate Heat Sink

   c. Spent Fuel Pool Cooling and Cleanup System

   d. Component Cooling Water System

   e. LOCTA_JR

   f. Anticipated Transients Without Scram

   Items 1.a through 1.d were previously provided by letter dated April 20, 2001. Items 1.e and 1.f have been approved and closed in SERs dated January 17, 2001 and March 16, 2001, respectively.

2. Identifies and evaluates the significant differences as they apply to WBN relative to the TPC topical report.

3. Provides confirmation of no adverse impact for the plant specific confirmatory checks recommended by the TPC topical report.

4. Provides evaluation of plant specific confirmatory checks that revealed an impact by TPBARs on reactor performance, plant systems and plant operations.

5. Addresses plant specific changes,

    a. Required Technical Specification (TS) changes for implementation and utilization of TPBARs at WBN.

6. Addresses other items cited in the SER, e.g.,

    a. TPBAR surveillance program.

    b. Lead Test Assembly (LTA) post irradiation results.

7. Provides additional information regarding the behavior of failed TPBARs during normal operation and during LBLOCA.

This report, the TPC topical reports (NDP-98-181, Revision 1, unclassified and non-proprietary version; NDP-98-153, classified and proprietary version), and the SER provide the basis for the TVA submittal that will request an amendment to WBN's operating license to allow irradiation of TPBARs. The proposed change is justified based on extensive analyses, testing, and evaluations of TPBARs documented in these reports. It has been determined that the proposed changes do not involve a significant hazards consideration and will have no significant environmental impact. In addition, it has been determined that the proposed changes will not endanger the health and safety of the public.

1 INTRODUCTION

1.1 PURPOSE OF PROGRAM

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

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

1.2 DESCRIPTION OF EFFORT

References 1 and 3 defined the plant specific evaluations and analyses required for WBN. Specifically, the scope of work for this report concentrated on:

1. Addressing the 17 plant specific interface issues listed in NUREG-1672, Section 5.1. The following interface items are being or have previously been submitted under a separate cover letter:

   a. Station Service Water

   b. Ultimate Heat Sink

   c. Spent Fuel Pool Cooling and Cleanup System

   d. Component Cooling Water System

   e. LOCTA_JR

   f. Anticipated Transients Without Scram

   Items 1.a through 1.d were previously provided by letter dated April 20, 2001. Items 1.e and 1.f have been approved and closed in SERs dated January 17, 2001 and March 16, 2001, respectively.

2. Identifying and evaluating the significant differences as they apply to WBN relative to Reference 1.

3. Providing confirmation of the plant specific confirmatory checks recommended by Reference 1.

4. Providing evaluation of plant specific confirmatory checks that revealed an impact by TPBARs on reactor performance, plant systems and plant operations.

5. Addressing plant specific changes,

   a. Required Technical Specification (TS) changes for implementation and utilization of TPBARs at WBN.

6. Addressing other items cited in Reference 3, e.g.,

   a. TPBAR surveillance program.

   b. Lead Test Assembly (LTA) post irradiation results.

7. Providing additional information regarding the behavior of failed TPBARs during normal operation and during LBLOCA.

1.3 WATTS BAR PLANT PARAMETERS

The TVA Watts Bar Unit 1 is a Westinghouse designed 4-loop pressurized water reactor with a rated thermal power of 3459 MW_t. Unit 1 contains 193 fuel assemblies of the 17x17 design. A fuel assembly consists of 264 fuel rods, 24 guide thimbles, and one instrumentation tube. Excess reactivity is typically controlled using 57 Hybrid Ag-In-Cd/B_4C rod cluster control assemblies (RCCA), burnable poison rod assemblies (BPRA), integral fuel burnable absorbers (IFBA), and soluble boron in the reactor coolant system (RCS).

Table 1-1 provides a comparison of Nuclear Steam Supply System (NSSS) parameters and features for the Reference Tritium Production Core (TPC) (Reference 1) and Watts Bar at a rated thermal power of 3459 MW_t. The Reference TPC was used as the basis for the TPBAR studies described in Reference 1. It was assumed that the Reference TPC was representative of candidate plants for the CLWR tritium program. Watts Bar's up-rated conditions (recently up-rated from 3411 MWt) were used as the basis for all evaluations and analyses described in this report.

Various key core design parameters are compared in Table 1-2 for the Reference TPC and Watts Bar at up-rated conditions. Watts Bar will insert TPBARs into the guide thimble locations of selected fuel assemblies to meet tritium production requirements. There will be no TPBARs in assemblies that are located under RCCAs or in assemblies containing source rods. Table 1-3 compares various key physical parameters between the Reference TPC and Watts Bar.

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

Table 1-1 NSSS Performance Parameters

	Reference TPC (Reference 1)	Watts Bar (Current)
Key Configuration Parameters:		
Number of Loops	4	4
Reactor Coolant Pump Motor, hp	7000	7000
Fuel Assembly	17x17 VANTAGE+	17x17 VANTAGE+
Containment Type	Dry	Ice
NSSS Performance Parameters:		
NSSS Power, MWt	3579	3475
Reactor Power, MWt	3565	3459
Thermal Design Flow, gpm/loop	93600	93100
Reactor Coolant Pressure, psia	2250	2250
Core Bypass Flow, %	8.4	9
Reactor Coolant Temperatures, °F		
Core Outlet	625.0	624.4
Vessel Outlet (T_{hot})	620.0	619.1
Core Average	593.0	592.8
Vessel Average	588.4	588.2
Vessel/Core Inlet (T_{cold})	556.8	557.3
Steam Generator Outlet	556.5	557.0
Steam Generator Performance		
Steam Temperature, °F	538.4	538.0
Steam Pressure, psia	950	947
Steam Flow, million lb/hr	15.92	15.36
Feedwater Temperature, °F	446	441.8
SG Maximum Tube Plugging, %	10	10

Table 1-2 Core Design Parameters			
Design Parameter	**Recent Watts Bar Design**	**Reference TPC Equilibrium Cycle**	**Watts Bar TPC Equilibrium Cycle**
Total Number of Feed Assemblies	76	140	96
Feed Loading (MTU)	35.0	59.2	44.4
Number of TPBARs	0	3344	2304
Total Grams of Tritium Produced	NA	2805	2065

Table 1-3 Key Physical Parameters	
Fuel assemblies in the core	193
Number of RCCAs	57
Fuel rods per assembly	264
Available guide thimble tubes per assembly	24
Active length of fuel, inches	144
Active length of TPBARs, inches	132

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

This report utilizes the TPC Topical Report (Reference 1) and the TPC SER (Reference 3) as the basis for the plant specific evaluations and analyses performed for Watts Bar. Extensive analyses, testing, and evaluations of TPBARs and their impact on a CLWR incorporating TPBARS were documented in Reference 1. It is the intent of this report not to reproduce the evaluations presented in Reference 1 that showed no impact of TPBAR utilization in a CLWR. However, each Standard Review Plan section in Reference 1 was reviewed to determine whether the "no impact" conclusion was valid for Watts Bar. Plant specific evaluations (and analyses if required) were performed for Watts Bar as recommended in Reference 1.

1.4.1 Watts Bar Report Sections Referencing the TPC Topical Report

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

It should also be noted that the numbering convention used in this report is identical to Reference 1 down to the third level. Sections 1 and 4 are exceptions to the above rule. Sections that appear to be missing have been purposely omitted because the information contained in Reference 1 is applicable to Watts Bar with TPBARs, the item for Watts Bar is addressed in Section 1.5 as an interface issue, or the specific evaluation or confirming check of the item is presented in Section 4, Table 4-1.

1.4.2 Differences Review

A review of References 1 and 3 was completed to identify any differences that exist between Watts Bar with TPBARs and the Reference TPC in Reference 1. In addition, the review included identifying any differences between the NRC conclusions documented in Reference 3 and Watts Bar with TPBARs. The noted differences are discussed in each section of this report, as appropriate. As part of this review, new information was identified concerning TPBAR performance following failures during normal plant operation and post-LBLOCA. This information is further discussed in Section 3.0.

Table 1-4 Summary of Standard Review Plan (SRP) Evaluations

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

Table 1-4 (cont.) Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Watts Bar Report Section
2.5.5	Stability of Slopes	2.2	No	NA
3.2.1	Seismic Classification	2.3	No	NA
3.2.2	System Quality Group Classification	2.3	No	NA
3.3.1	Wind Loadings	2.3	No	NA
3.3.2	Tornado Loadings	2.3	No	NA
3.4.1	Flood Protection	2.3	No	NA
3.4.2	Analysis Procedures	2.3	No	NA
3.5.1.1-3.5.1.6	Missiles	2.3	No	NA
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2.3	No	NA
3.5.3	Barrier Design Procedures	2.3	No	NA
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	2.3	No	NA
3.6.2	Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	2.3	No	NA
3.7.1	Seismic Design Parameters	2.3	No	NA
3.7.2 3.7.3	Seismic System and Subsystem Analysis	2.3	No	NA
3.7.4	Seismic Instrumentation	2.3	No	NA
3.8.1 3.8.2	Concrete Containment/Steel Containment	2.3	No	NA
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	2.3	No	NA
3.8.4	Other Seismic Category 1 Structures	2.3	No	NA
3.8.5	Foundations	2.3	No	NA
3.9.1	Special Topics for Mechanical Components	2.3	Yes	Sec. 4, Table 4-1
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	2.3	Yes	Sec. 4, Table 4-1
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	2.3	Yes	Sec. 4, Table 4-1

Table 1-4 (cont.) Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Watts Bar Report Section
3.9.4	Control Rod Drive Systems	2.3	Yes	Sec. 4, Table 4-1
3.9.5	Reactor Pressure Vessel Internals	2.3	Yes	Sec. 4, Table 4-1
3.9.6	Inservice Testing of Pumps and Valves	2.3	No	NA
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	2.3	No	NA
3.11	Environmental Qualification of Mechanical and Electrical Equipment	2.3	Yes	Sec. 4, Table 4-1
4.2	Fuel System Design	2.4	Yes	2.4.2
4.3	Nuclear Design	2.4	Yes	2.4.3
4.4	Thermal and Hydraulic Design	2.4	Yes	2.4.4
4.5.1	Control Rod Drive Structural Materials	2.4	No	NA
4.5.2	Reactor Internal and Core Support Materials	2.4	No	NA
4.6	Functional Design of Control Rod Drive System	2.4	Yes	Sec. 4, Table 4-1
5.2.1.1 5.2.1.2	Compliance with the Codes and Standards Rule, 10CFR50.55a and Applicable Code Cases	2.5	No	NA
5.2.2	Overpressurization Protection	2.5	Yes	Sec. 4, Table 4-1
5.2.3	Reactor Coolant Pressure Boundary Materials	2.5	No	NA
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	2.5	No	NA
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	2.5	No	NA
5.3.1	Reactor Vessel Materials	2.5	Yes	1.5.4
5.3.2	Pressure-Temperature Limits	2.5	Yes	1.5.4
5.3.3	Reactor Vessel Integrity	2.5	Yes	1.5.4
5.4.1.1	Pump Flywheel Integrity (PWR)	2.5	No	NA
5.4.2.1	Steam Generator Materials	2.5	No	NA
5.4.2.2	Steam Generator Tube Inservice Inspection	2.5	No	NA
5.4.7	Residual Heat Removal (RHR) System	2.5	Yes	Sec. 4, Table 4-1

Table 1-4 (cont.) Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Watts Bar Report Section
5.4.11	Pressurizer Relief Tank	2.5	No	NA
5.4.12	Reactor Coolant System High Point Vents	2.5	No	NA
6.1.1	Engineered Safety Features Materials	2.6	No	NA
6.1.2	Protective Coating Systems (Paints) – Organic Materials	2.6	Yes	Sec. 4, Table 4-1
6.2.1	Containment Functional Design	2.6	Yes	Sec. 4, Table 4-1, 6.2.1
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	2.6	No	NA
6.2.1.1.B	Ice Condenser Containments	2.6	No	NA
6.2.1.2	Subcompartment Analysis	2.6	No	NA
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	2.6	Yes	Sec. 4, Table 4-1, 6.2.1
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	2.6	Yes	Sec. 4, Table 4-1, 6.2.1
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2.6	Yes	Sec. 4, Table 4-1, 6.2.1
6.2.2	Containment Heat Removal Systems	2.6	Yes	Sec. 4, Table 4-1
6.2.3	Secondary Containment Functional Design	2.6	No	NA
6.2.4	Containment Isolation System	2.6	No	NA
6.2.5	Combustible Gas Control in Containment	2.6	Yes	1.5.6
6.2.6	Containment Leakage Testing	2.6	No	NA
6.2.7	Fracture Prevention of Containment Pressure Boundary	2.6	No	NA
6.3	Emergency Core Cooling System	2.6	Yes	Sec. 4, Table 4-1
6.4	Control Room Habitability Systems	2.6	Yes	1.5.5
6.5.1	ESF Atmosphere Cleanup Systems	2.6	No	NA
6.5.2	Containment Spray as a Fission Product Cleanup System	2.6	No	NA
6.5.3	Fission Product Control Systems and Structures	2.6	Yes	Sec. 4, Table 4-1

Table 1-4 (cont.) Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Watts Bar Report Section
6.5.4	Ice Condenser as a Fission Product Cleanup System	2.6	No	NA
6.6	Inservice Inspection of Class 2 and 3 Components	2.6	No	NA
7.1	Instrumentation and Controls-Introduction	2.7	No	NA
7.2	Reactor Trip System	2.7	Yes	Sec. 4, Table 4-1
7.3	Engineered Safety Features Systems	2.7	Yes	Sec. 4, Table 4-1
7.4	Systems Required for Safe Shutdown	2.7	Yes	Sec. 4, Table 4-1
7.5	Information Systems Important to Safety	2.7	Yes	Sec. 4, Table 4-1
7.6	Interlock Systems Important to Safety	2.7	No	NA
7.7	Control Systems	2.7	Yes	Sec. 4, Table 4-1
8.0	Electric Power	2.8	Yes	Sec. 4, Table 4-1
9.1.1	New Fuel Storage	2.9	Yes	1.5.10
9.1.2	Spent Fuel Storage	2.9	Yes	1.5.10
9.1.3	Spent Fuel Pool Cooling and Cleanup System	2.9	Yes	1.5.11
9.1.4	Light Load Handling System	2.9	Yes	1.5.7
9.1.5	Overhead Heavy Load Handling Systems	2.9	Yes	2.9.1.1
9.2.1	Station Service Water System	2.9	Yes	1.5.8
9.2.2	Reactor Auxiliary Cooling Water Systems	2.9	Yes	1.5.12
9.2.3	Demineralized Water Makeup System	2.9	Yes	1.5.13
9.2.4	Potable and Sanitary Water Systems	2.9	No	NA
9.2.5	Ultimate Heat Sink	2.9	Yes	1.5.9
9.2.6	Condensate Storage Facilities	2.9	No	NA
9.3.1	Compressed Air System	2.9	No	NA
9.3.2	Process and Post-Accident Sampling Systems	2.9	Yes	2.9.6
9.3.3	Equipment and Floor Drainage System	2.9	No	NA
9.3.4	Chemical and Volume Control System	2.9	Yes	2.9.1.2
10.0	Steam and Power Conversion System	2.10	Yes	Sec. 4, Table 4-1

Table 1-4 (cont.)	Summary of Standard Review Plan (SRP) Evaluations			
SRP Section Number	**SRP Section Title**	**NDP-98-181 Revision 1 Section**	**Plant Specific Evaluation Needed**	**Watts Bar Report Section**
11.1	Source Terms	2.11	Yes	2.11.2
11.2	Liquid Waste Management Systems	2.11	Yes	1.5.14 and 2.11.3
11.3	Gaseous Waste Management Systems	2.11	Yes	2.11.4
11.4	Solid Waste Management Systems	2.11	Yes	2.11.5
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	2.11	Yes	1.5.15 and 2.11.6
12.1	Assuring that Occupational Radiation Exposures are As Low As is Reasonably Achievable (ALARA)	2.12	No	NA
12.2	Radiation Sources	2.12	Yes	2.12.2
12.3-12.4	Radiation Protection Design Features	2.12	Yes	2.12.3
12.5	Operational Radiation Protection Program	2.12	Yes	2.12.4
13.1.1	Management and Technical Support Organization	2.13	No	NA
13.1.2-13.1.3	Operating Organization	2.13	No	NA
13.2.1-13.2.2	Training	2.13	Yes	2.13.1.1
13.3	Emergency Planning	2.13	Yes	2.13.1.2
13.4	Operation Review	2.13	No	NA
13.5.1-13.5.2	Administrative, Operating, and Maintenance Procedures	2.13	Yes	2.13.1.3
13.6	Physical Security	2.13	Yes	2.13.2
14.2	Initial Plant Test Program-Final Safety Analysis Report	2.14	Yes	2.14.2
15.1.1-15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	2.15	Yes	Sec. 4, Table 4-1
15.1.5	Steam System Piping Failures Inside and Outside of Containment	2.15	Yes	Sec. 4, Table 4-1
15.1.5, Appendix A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	2.15	Yes	2.15.6.4
15.2.1-15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve, and Steam Pressure Regulator Failure (Closed)	2.15	Yes	Sec. 4, Table 4-1

Table 1-4 (cont.) Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Watts Bar Report Section
15.2.6	Loss of Non-emergency AC Power to the Station Auxiliaries	2.15	Yes	Sec. 4, Table 4-1
15.2.7	Loss of Normal Feedwater Flow	2.15	Yes	Sec. 4, Table 4-1
15.2.8	Feedwater System Pipe Breaks Inside and Outside of Containment	2.15	Yes	Sec. 4, Table 4-1
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	2.15	Yes	Sec. 4, Table 4-1
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2.15	Yes	Sec. 4, Table 4-1
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition	2.15	Yes	Sec. 4, Table 4-1
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2.15	Yes	Sec. 4, Table 4-1
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2.15	Yes	Sec. 4, Table 4-1
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	2.15	Yes	Sec. 4, Table 4-1
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	2.15	Yes	Sec. 4, Table 4-1
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	2.15	Yes	Sec. 4, Table 4-1
15.4.8	Spectrum of Rod Ejection Accidents	2.15	Yes	Sec. 4, Table 4-1
15.4.8, Appendix A	Radiological Consequences of a Control Rod Ejection Accident	2.15	Yes	2.15.6.7
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	2.15	Yes	2.15.2.1 and Sec. 4, Table 4-1
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	2.15	Yes	Sec. 4, Table 4-1

Table 1-4 (cont.) Summary of Standard Review Plan (SRP) Evaluations

SRP Section Number	SRP Section Title	NDP-98-181 Revision 1 Section	Plant Specific Evaluation Needed	Watts Bar Report Section
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2.15	Yes	2.15.6.9
15.6.3	Radiological Consequences of Steam Generator Tube Failure	2.15	Yes	2.15.6.5
15.6.5 and Appendices A & B	Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	2.15	Yes	2.15.5 and 2.15.6.3
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	2.15	Yes	2.11.3
15.7.4	Radiological Consequences of Fuel Handling Accidents	2.15	Yes	2.15.6.6
15.7.5	Spent Fuel Cask Drop Accidents	2.15	Yes	Sec. 4, Table 4-1
15.8	Anticipated Transients Without Scram (ATWS)	2.15	Yes	1.5.17
16.0	Technical Specifications	2.16	Yes	1.6
17.1	Quality Assurance During the Design and Construction Phases	2.17	Yes	1.5.2 and 2.17
17.2	Quality Assurance During the Operations Phase	2.17	Yes	1.5.2 and 2.17
17.3	Quality Assurance Program Description	2.17	No	NA
18.1	Control Room	2.18	No	NA
18.2	Safety Parameter Display System (SPDS)	2.18	No	NA

1.5 WATTS BAR PLANT SPECIFIC INTERFACE ISSUES

During the NRC's review of Reference 1, the NRC determined there are certain plant specific interface issues for which the licensee must submit additional information and/or analyses. This information would be used to support a plant specific license amendment to the facility's operating license for authorization to operate a tritium production core. Each specific interface issue has been evaluated for Watts Bar and is discussed below. Note that submittals (References 4 and 5) to the NRC have been made to address the items in Sections 1.5.16 and 1.5.17. A separate submittal dated April 20, 2001 addresses the items in Sections 1.5.8, 1.5.9, 1.5.11, and 1.5.12.

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

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

1.5.1 Handling of TPBARs

NUREG-1672, Section 1.3, "DOE did not address the activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment because these activities are dependent on the fuel pool design. Therefore, the staff has identified this as an interface item that must be *addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

NUREG-1672, Section 2.9.2, "In addition, DOE did not address the activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment because these activities are dependent on the fuel pool design. Therefore, the staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

NUREG-1672, Section 3.7, "DOE has described the consequences of potential handling damage resulting from refueling operations and during onsite fuel assembly movement and handling with TPBARs installed. If an irradiated TPBAR is breached as a result of mishandling in the spent fuel pool, only a small fraction of the tritium inventory would be released. The tritium in the open pores of the pellet (tens of Ci) will be released when water comes in contact with the pellet. Further release may occur gradually due to the limited leaching of the pellets and would provide adequate time to isolate the damaged TPBAR cluster to prevent further release into the pool. DOE did not address post-irradiation movement of the TPBARs outside of fuel assemblies. Therefore, the staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

TVA has completed a preliminary design of a TPBAR Consolidation Fixture (TCF) to be installed in the cask loading pit for consolidation activities (see Figures 1.5.1-1 and 1.5.1-2). The TCF is quality related in accordance with TVA's NRC accepted QA Program. It will normally be stored in the cask lay-down area when not in use. The TCF fixture includes a video monitoring system, lighting, and tools designed to remove TPBARs from its baseplate. The TPBARs are deposited into a consolidation canister (see Figure 1.5.1-3) (up to 300 TPBARs per canister). The loaded canister is transferred back into the spent fuel pool for short term storage until ultimately being placed into shipping casks for transport off-site to DOE.

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

Consolidation Sequence

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

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

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

After TPBARs have been removed from a baseplate, the baseplate and any attached thimble plugs will be removed from the fixture, and placed in storage. The process is repeated until the canister is filled with up to 300 TPBARs. Disposal or storage of the baseplates and thimble plugs will be in accordance with accepted radwaste programs.

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

Subsequently, a shipping cask is placed into the cask loading pit. The cask is handled by the Auxiliary Building crane in accordance with NUREG-0612 program requirements. The canisters are transferred into the submerged cask. The cask is removed from the cask loading pit, drained of water and decontaminated, packaged and certified for shipment. This shipping process is repeated until all TPBARs irradiated during the past operating cycle have been shipped. The consolidation process is based upon accepted industry practices. The evolutions are performed with sufficient shielding to minimize exposure, and specialized tooling has been developed to streamline the process.

The consequences of a breached TPBAR as a result of mishandling in the spent fuel pool are addressed in Section 2.15.6.6.

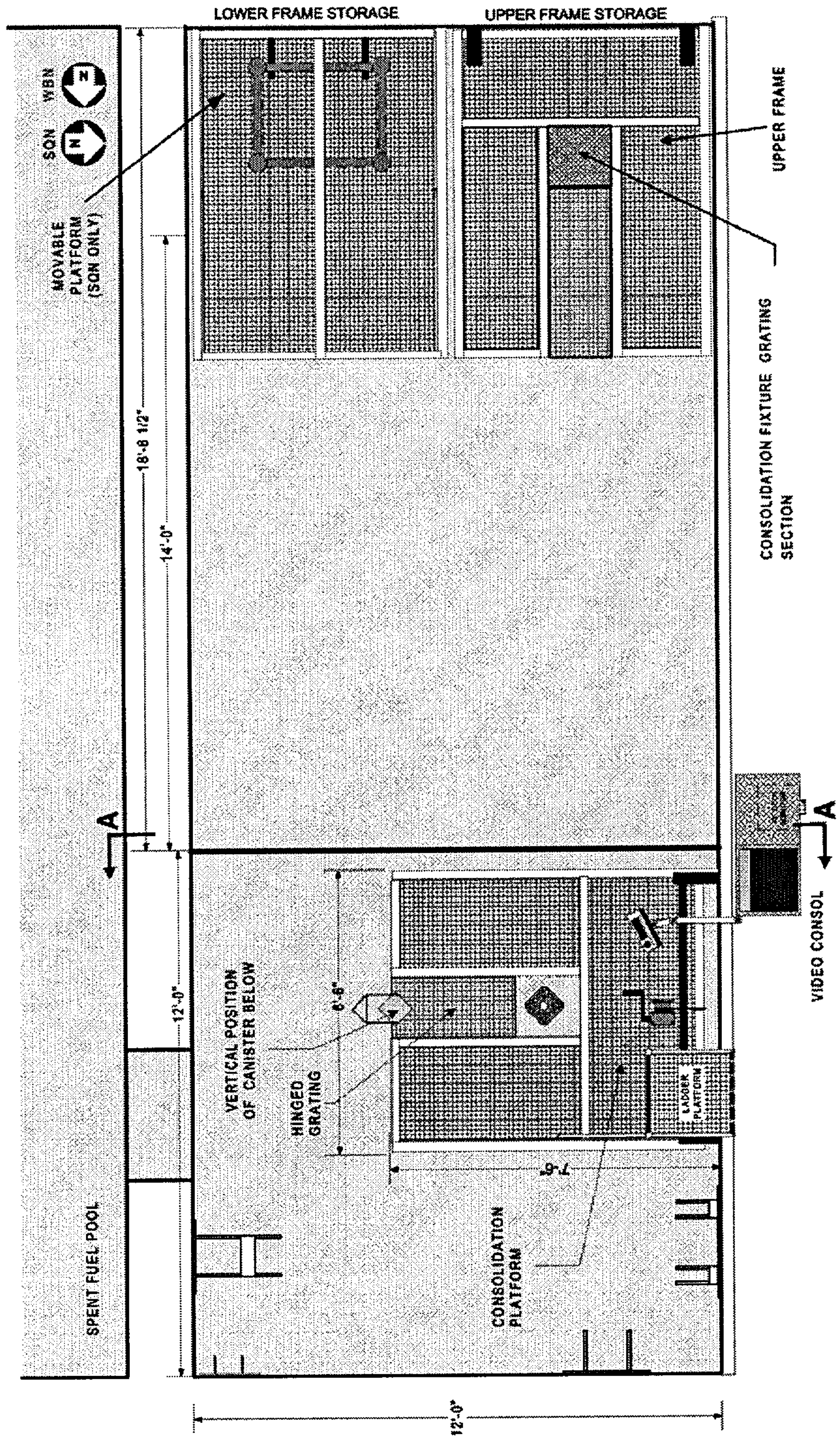


Figure 1.5.1-1
Consolidation Plan View

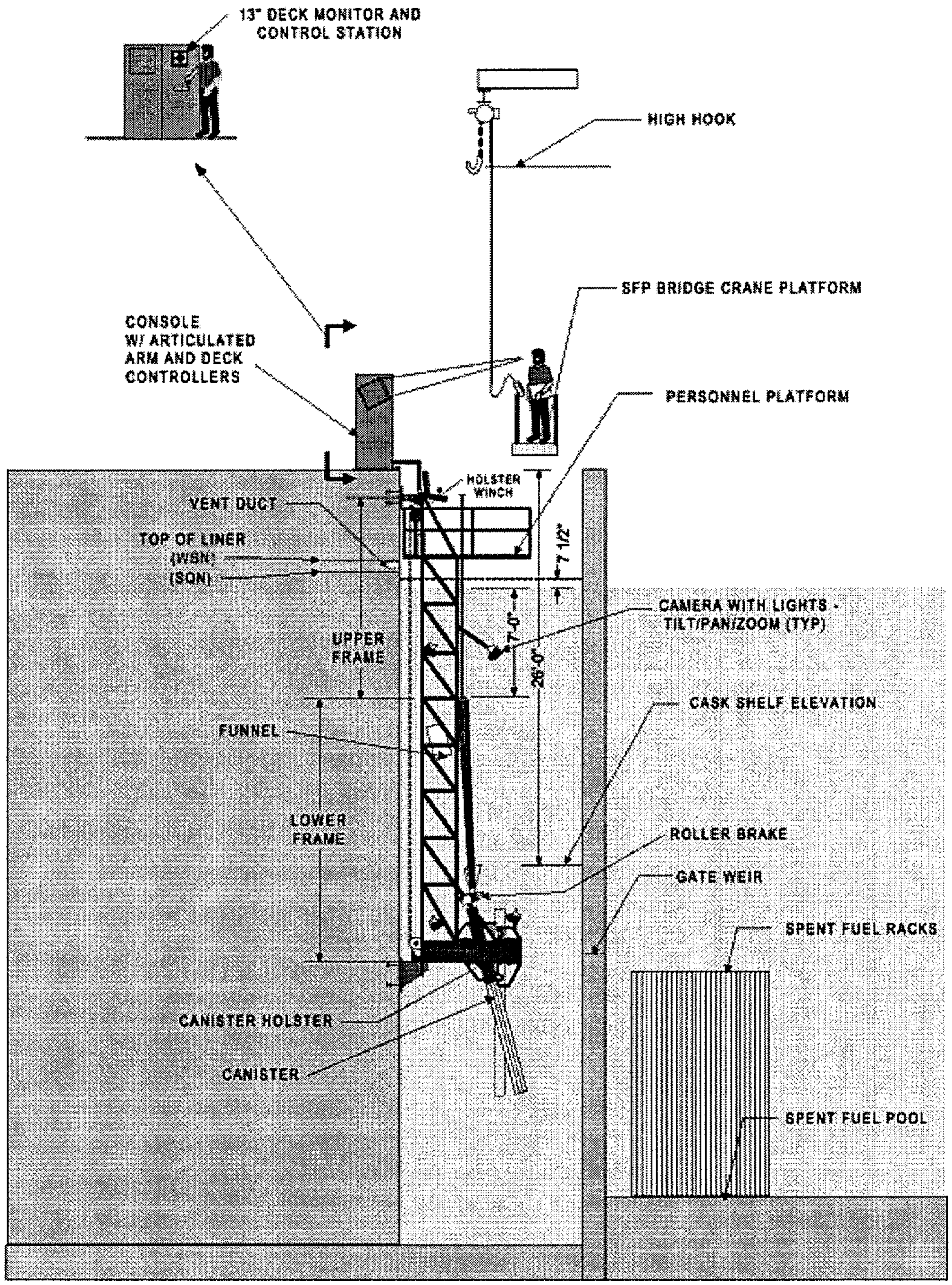
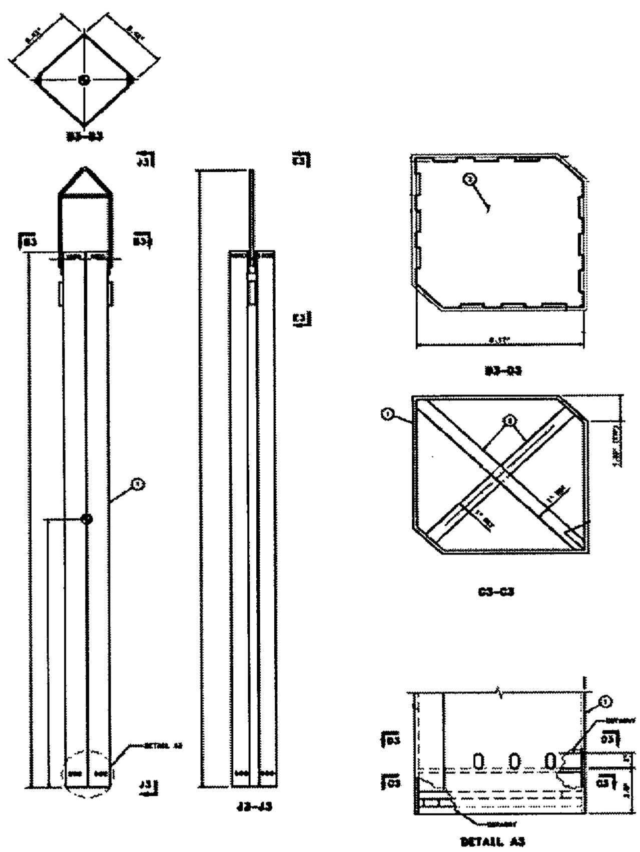
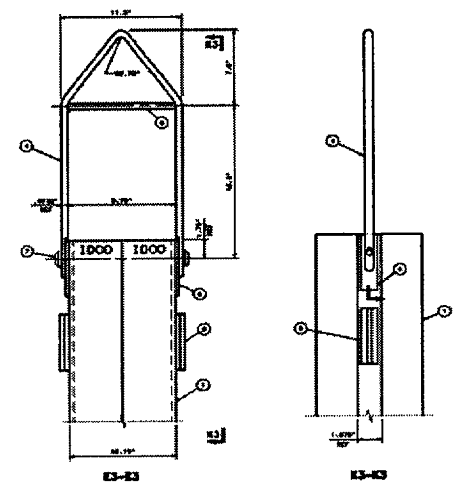


Figure 1.5.1-2
Consolidation Layout





1. Each canister shall have a unique identification clearly denoted on four locations of the canister at the top as shown (ID00). Lettering is to be a minimum 3/4" tall etched, engraved, or stamped on metal.

2. Total dry weight: 904 lb (Including 300 TPBARs)
   Dry weight empty: 184 lb

1-Canister	5-Bail crossmember
2-Energy transfer bars	6-Hinge plate
3-Canister bottom plate	7-Hinge pin
4-Bail	8-Lifting lug

Figure 1.5.1-3
Consolidation Canister

1.5.2 Procurement and Fabrication Issues

NUREG-1672, Section 1.3 "Independent of its review of the DOE TPC topical report, the staff is conducting vendor-related activities with respect to quality assurance (QA) plans and fabrication inspections in order to determine compliance with the requirements of Appendix B to 10 CFR Part 50 and with 10 CFR Part 21. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

NUREG-1672, Section 2.17.1 "DOE has not yet selected the supplier for the fabrication of the production core TPBARs, and NRC review and inspection of supplier/vendor QA programs is not within the scope of this evaluation. Procurement processes performed on behalf of DOE for production core TPBAR components by contractors other than the production core TPBAR fabricator will also be subject to NRC review and inspection. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant specific application for authorization to irradiate TPBARs for the production of tritium."

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

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

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

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

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

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

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

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

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

1.5.3 Compliance with DNB Criterion

> *NUREG-1672, Section 2.4.4, "DOE's analyses regarding the incorporation of the TPBARs in the reference plant showed that the bypass flow will remain within its design limit of 8.4 percent, and that the DNB criterion will continue to be met with no feature of the TPBAR component affecting the coolability of the core. The staff agrees with this assessment. However, the continued compliance with the DNB criterion, given the operating conditions of a particular plant, must be evaluated. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

During its review of the DOE TPC topical report, the NRC staff identified compliance with the Departure from Nucleate Boiling (DNB) criterion as an interface issue for which plant-specific information would be required in the licensee's submittal to support an amendment to the facility operating license for authorization to operate a tritium production core. This criterion requires the demonstration that DNB will not occur on the most limiting fuel rod on at least a 95 percent probability at a 95 percent confidence level. For the Watts Bar Tritium Production equilibrium cycle, the normal Thermal/Hydraulic DNB related reload analyses were performed using VIPRE-01 (Reference 6) and are described in more detail in Section 2.4.4. The following detailed Thermal Hydraulic evaluations were performed.

1. An axial power shape study was performed to assure that the limiting power distributions used in design would still be valid in the presence of the TPBAR. This compares power shapes resulting from depletion during operation of the cycles to reference shapes used as the basis for thermal/hydraulic design analyses.

2. The Steamline Break with Rod Withdrawal at Power transient was analyzed to demonstrate the continued acceptability of the Departure from Nucleate Boiling Ratio (DNBR) design basis for this transient.

3. Zero Power Hypothetical Steamline Break was analyzed to demonstrate that the DNBR design basis was met.

The axial power shape comparison showed that with the assumption of the current operation strategy, the reference power shapes assumed in the current safety analysis for Watts Bar would remain bounding. The TPBAR would not present any excessive power distribution changes beyond those, which are already bounded within the thermal/hydraulic design bases. In addition, the results of the DNB analyses showed that the DNBR design basis was met. Therefore, the presence of TPBARs in the reload core design did not challenge the DNB criterion. An explicit check of the DNB criterion is included in the cycle-specific reload safety evaluation performed for each Watts Bar reload core. Continued performance of this check will validate the acceptability of each reload core for operation within the DNB design limits.

1.5.4 Reactor Vessel Integrity Analysis

> *NUREG-1672, Section 2.5.3, "The TPC topical report identifies the applicable regulations and describes methods for demonstrating compliance with Appendices G and H to 10 CFR Part 50 and with 10 CFR 50.61. In the TPC topical report, DOE concludes, and the staff agrees, that the reference plant's pressure/temperature limits report (PTLR) and final safety analysis report (FSAR) would need to be updated to reflect the change to the PTS value and include the updated P-T curves for the applicable EFPYs. In addition, because the reactor vessel integrity analyses are dependent upon the plant-specific materials properties and neutron fluence, the staff concludes that a licensee participating in DOE's program for the CLWR production of tritium must present the material properties for its reactor vessel and perform analyses that demonstrate it will meet the requirements of Appendices G and H to 10 CFR Part 50 and of 10 CFR 50.61. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

Several analyses are performed to determine the impact that neutron irradiation has on the Watts Bar (WBN) Reactor Vessel (RV) integrity. These analyses include a surveillance capsule withdrawal schedule, heatup and cooldown pressure-temperature limit curves, pressurized thermal shock calculations and upper shelf energy evaluations. All of these analyses and evaluations can be affected by changes in the neutron fluences and operating temperatures and pressures. The evaluation of the Tritium production core includes the recent 1.4% power uprate.

The most critical area is the beltline region of the RV since it is predicted to be most susceptible to neutron damage. The beltline region is defined in ASTM E185-82 (Reference 7) as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material."

<u>Input Parameters and Assumptions</u>

With regard to the neutron flux that impinges on the reactor pressure vessel (RPV), the existing fast ($E > 1.0$ MeV) neutron fluence projections bound the corresponding projections for the tritium production core design. The existing fast neutron RPV exposures were originally generated after the first operating cycle was completed, and are based on exposure rates resulting from conservative loading pattern assumptions rather than the actual low leakage loading pattern used in the conventional (i.e., non-tritium producing) Watts Bar core design. In a typical low leakage loading pattern, the assemblies on the periphery are mostly low reactivity, twice-burned assemblies that naturally operate at very low powers. This kind of loading pattern limits the accumulation of fluence on the reactor vessel. Because maximum tritium production will result in a larger feed region (up to 96 assemblies used in the example equilibrium cycle tritium production core), the burned assemblies placed on the core periphery are only once-burned and therefore more reactive. To mitigate the potential impact this would have on the vessel fluences and consequently vessel lifetime, the equilibrium cycle tritium production core inserts TPBARs in key locations on the periphery to reduce the powers in specific assemblies important to peak vessel fluence. The equilibrium core design, therefore, places clusters of 16 TPBARs in the sixteen corner fuel assembly locations (see Figure 1.5.4-1). While these TPBARs produce only a modest amount of tritium, they are effective in reducing the power in these locations to an amount comparable to the current Watts Bar low leakage loading pattern. The actual Tritium Production Core implementation may involve a lower number of feed assemblies; however, the cycle specific core designs will employ the approach of maintaining the power in critical peripheral assemblies such that the existing design-basis RPV exposure projections remain bounding. This approach may include the use of a minimum number of TPBARs, or other discrete burnable absorber rods inserted into the critical peripheral assemblies, as required.

<u>Applicable Analyses</u>

Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions. Since the fluence projections used in development of the current withdrawal schedules remain bounding, the current withdrawal schedules remain valid for the tritium production core designs.

Heat-up and Cooldown Pressure - Temperature Limit Curves

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

Since the fluence projections used in developing the ART values remain bounding, the current heatup and cooldown curves remain valid for the tritium production core design.

Pressurized Thermal Shock (PTS)

The current RT_{PTS} values for Watts Bar do not exceed the screening criteria of the PTS Rule. Since the fluence projections used in developing the RT_{PTS} values remain bounding, the existing RT_{PTS} values remain valid for the tritium production.

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits (Reference 8) were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at end of life. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The current peak inside surface RT_{NDT} values at EOL and license renewal were calculated to be 253°F and 265°F, respectively. Since the fluence projections used in development of the current peak inside surface RT_{NDT} values at EOL remain bounding, these RT_{NDT} values remain valid for the tritium production core and the applicable ERG category will not change.

Upper Shelf Energy (USE)

The current neutron fluence projections remain bounding for the tritium production core. Therefore, the current USE values for Watts Bar remain applicable for the tritium production core.

<u>Conclusions</u>

It is concluded that the tritium production core will not have a significant impact on the Watts Bar reactor vessel based on the following:

1. The core design employs power suppression techniques in key peripheral fuel assembly locations so that the power in those locations remains comparable to the current Watts Bar low leakage loading patterns.

2. The fluence projections for the tritium production core are bounded by the existing fluence projections for Watts Bar.

3. Therefore, the existing RV integrity analyses remain valid for the Tritium Program.

	1	2	3	4	5	6	7	8
1								
2								
3								
4								16 TPBARs
5								
6							16 TPBARs	
7						16 TPBARs		
8				16 TPBARs				

Figure 1.5.4-1
Location of TPBAR Assemblies Used for Suppressing Neutron Fluence on Watts Bar Vessel Wall in Example Equilibrium Cycle

1.5.5 Control Room Habitability Systems

NUREG-1672, Section 2.6.1, "Therefore, the staff concludes that, except for the dose criteria issue, the TPC topical report adequately addresses this matter, but that a plant-specific assessment will be needed. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to produce tritium for DOE."

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

An analysis was performed for Watts Bar Nuclear Plant to determine the control room operator dose due to an ECCS leak outside of containment following a LOCA. This analysis was performed for a conventional core and for a Tritium Production Core. In both cases the latest version of COROD (R5) was utilized and the Whole Body, Skin, and Thyroid doses were based on Federal Guideline Reports (FGR-11 and FGR-12) dose conversion factors. The TEDE is also determined. The analyses also incorporated new dispersion factors with X/Q factors determined by NRC approved code ARCON96. The ECCS leakage outside of containment was assumed to be 3,760 cc/hr. This is the same value used for previous Watts Bar analyses and is conservative considering the fact that the actual calculated value is approximately 334 cc/hr, which is less than ten percent of the assumed value.

The specific results of the analyses are provided in Table 2.15.6-2. These analyses and the summary data presented on Table 2.15.6-2 demonstrate that the potential increase in dose resulting from use of TPBARs is within the prescribed regulatory limits. Control room habitability requirements continue to be met for 10CFR50 Appendix A, GDC 19.

1.5.6 Specific Assessment of Hydrogen Source and Timing of Recombiner Operation

NUREG-1672, Section 2.6.2, "The staff agrees with the DOE conclusions, based on the conservative assessment of the TPBARs on the combustible gas concentrations in containment following a LOCA, that the combustible gas control systems are not expected to be affected by the TPC. However, the staff concludes that a plant-specific assessment is required to quantify the sources and to determine the time at which initiation of recombiner operation should commence to limit the hydrogen concentration to acceptable levels. The staff has identified this as an *interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

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

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

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

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

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

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

TPBAR Metal-Water Reaction

One of the potential sources of hydrogen unique to a TPC design is that associated with zirconium getter materials contained within the TPBARs. The zirconium that is subject to the zirconium-water reaction is specified in 10 CFR 50.44 (Reference 9) to be only that associated with the "... fuel cladding surrounding the active fuel region ..." and "... the mass of metal in the cladding cylinders surrounding the fuel ...." (Note: the Watts Bar evaluation conservatively assumes the grid spacers are also subject to the reaction). This follows since it is generally only the metal in the active core region that is subject to the high temperatures (in excess of 1800°F), which are necessary for the zirconium-water reaction to occur. However, if the TPBAR cladding is breached following a LBLOCA, the potential for a metal water reaction with internal zirconium components can be postulated.

Based on the chemical stoichiometry of the zirconium-water reaction, one pound-mole of zirconium metal reacted must produce two pound-moles of hydrogen. That is, 7.9 standard cubic feet (scf) of hydrogen gas is produced for each pound of zirconium metal reacted. The maximum amount of zirconium associated with the getter material (300 grams per TPBAR) in 2,304 TPBARs (i.e., the total number of TPBARs in an equilibrium cycle in Watts Bar) is 1524 pounds.

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

$$F = 12 \text{ in}/132 \text{ in} = 0.0909,$$

where a TPBAR absorber length of 132 inches is used in order to conservatively estimate the fraction. The value determined above is equal to the fraction of the total TPBAR zirconium mass involved in the reaction. Then, the equivalent hydrogen that could be released is

$$V' = 1524 \times 0.0909 \times 7.9 = 1094 \text{ scf.}$$

TPBAR Tritium and Hydrogen Inventories

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

Conservatively assuming the design limit of 1.2 grams per rod at the end of the fuel cycle, the equivalent volume of tritium gas (T_2) associated with the mass of tritium contained within the 2304 TPBARs in the core is 364 scf of T_2.

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

Results and Conclusions

The additional hydrogen inventories that are conservatively estimated to be associated with a TPC are 1094 scf associated with zirconium-water reactions with the TPBAR getter materials, 364 scf of tritium gas from the TPBARs, and 16 scf of hydrogen from $^3He(n,p)T$ reactions inside the rods. This sums to a total of 1474 scf as the potential additional amount of hydrogen contributed by the TPBARs following a LBLOCA.

This inventory would be expected to exist in the primary coolant as water or tritiated water (HTO or T_2O), rather than as a gas. However, even if the complete hydrogen/tritium inventory associated with a TPC is conservatively assumed to be released to the containment atmosphere as gas, the added inventory represents only a 4% increase in the amount of hydrogen gas in the containment one day after a

LBLOCA. That is, the total inventory in the containment at one day after a LBLOCA, including TPC sources, is 39,225 scf which is 4% higher than the value of 37,751 scf calculated on the basis of operation with a conventional core.

The lower flammability limit for hydrogen in the containment atmosphere that should not be exceeded as defined in USNRC Regulatory Guide 1.7 (Reference 10) is 4 volume percent. For the Watts Bar plant with a total containment free volume of 1,230,000 ft^3 a concentration of 4 volume percent equates to approximately 49,200 scf of hydrogen. Thus, the contribution of the TPC tritium inventory to the amount of hydrogen associated with the recommended Regulatory Guide limit is only about 3%, i.e.,

$$F' = 1474/49{,}200 = 0.030.$$

It is concluded that even based on highly conservative assumptions, the TPBARs are not a significant contributor to the post-LOCA hydrogen inventory. The TPC will not have a significant impact on the total hydrogen production and concentrations within the containment, as compared to the values associated with operation with a conventional core. The maximum hydrogen concentration with a TPC can be maintained at less than the lower flammability limit of 4 volume percent, with one recombination train in operation, if the recombiner is started at a containment hydrogen concentration of 3 volume percent following the accident. This is the same initiation time as described in the Watts Bar UFSAR.

1.5.7 Light – Load Handling System

> *NUREG-1672, Section 2.9.1, "DOE evaluated the affect of TPBARs on the light load handling system for the reference plant against the guidance of SRP Section 9.1.4. DOE states, and the staff agrees, that the incorporation of the TPBARs has no effect on this system. However, DOE concludes, and the staff agrees, that because of the increase in weight of TPBARs compared to burnable poison rod assemblies, this effect should be evaluated on a plant-specific basis. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

The TPBAR consolidation and shipping phase of the program has been evaluated with respect to the light load handling system.

The handling of items during TPBAR consolidation will be performed by using the Spent Fuel Pit Bridge Crane which utilizes a specialized fixture and tooling to transport the TPBAR assemblies, consolidate individual rods into consolidation canisters, dispose of empty baseplates, transport the canisters for storage in the Spent Fuel Pit, and finally load canisters into shipping casks for transport off-site.

The weight of a fuel assembly containing 24 TPBARs (including the holddown assembly) is less than a fuel assembly with a RCCA and therefore is bounded by the current assumed weight of assembly for purposes of analyzing fuel handling and storage facilities. The fuel assembly with TPBARs has the same external configuration as a fuel assembly without TPBARs allowing for interface with existing fuel handling/storage equipment. Additionally, this weight is conservative for purposes of defining a NUREG-0612 "Heavy Load."

During consolidation of TPBARs from a baseplate, rods are released from the baseplate one at a time. (For a description of the consolidation process, see Section 1.5.1.) Additionally, the consolidation fixture is designed to seismic category 1(L) to preclude damage to consolidated TPBARs while in the fixture and to the spent fuel pool liner. After approximately 300 rods are released into a canister, the loaded canister is transported to a designated spent fuel pool cell location using a canister handling tool suspended from the SFP Bridge Crane. Handling of the loaded canister with the following analysis/design features will limit, to an acceptable level, the possibility of damage to more than 24 TPBARs during handling:

1. In accordance with NUREG-0612, -0554 and ANSI N14.6, the Spent Fuel Pit Bridge Crane and canister lifting device will contain sufficient aspects of the single failure proof criteria to preclude a drop of the loaded canister as delineated below.

   - The SFP Bridge Crane is equivalent single failure proof with respect to structural integrity in accordance with NUREG-0612 (NUREG-0554) due to the following:

     a. Since the SFP Bridge Crane has a capacity of 4000 lbs and the weight of the submerged loaded canister is approximately 700 lbs, the crane has safety factors twice the normally required values.

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

   - The lifting tool is provided with a safety lanyard to limit canister descent in the fuel pool to such an extent that spilling of the TPBARs out of the open topped canister, if the canister bottom were to hit an obstruction and cause the canister to tip, is prevented. The lanyard is sized to stop the canister from a maximum hook speed of 40 feet per minute. Administrative requirements require that the safety lanyard be attached to the lifting tool when the canister is not engaged in a SFP rack cell, the consolidation fixture holster, or cask by at least 12 inches.

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

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

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

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

1.5.8 Station Service Water System

NUREG-1672, Section 2.9.1, "The staff has reviewed the information presented by DOE and concludes that the effect on the SSWS is not safety significant, because the additional heat load introduced by TPBARs is very low and is indirectly transferred to the SSWS. The staff also agrees that, during the generic review of the TPC topical report, a quantitative analysis of the effect of the TPBARs on the SSWS was not appropriate. However, DOE concludes, and the staff agrees, that a quantitative analysis for the SSWS needs to be addressed by licensees participating in DOE's program for the CLWR production of tritium. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

NOTE: This item is being submitted separately.

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

Tritium Impact on Spent Fuel Pool Decay Heat

TVA has prepared a quantitative analysis of expected spent fuel decay heat for both Tritium Production Cores (TPCs) and non-TPCs. The analysis is based on comparative decay heat data prepared by TVA for a base non-tritium core, a TPC with 80 fresh fuel assemblies (80-feed), and a TPC with 96 fresh fuel assemblies (96-feed). The results of the analysis show that the 80 feed case was limiting for decay heat (i.e., freshly offloaded core), and the 80-feed TPC core contributes a slightly higher decay heat over the non-TPC and the 96-feed TPC, due to isotopic composition differences between the base and TPC cores, for the same design basis reactor power level. The results of the analysis show that the 96-feed case was limiting for residual SFP heat (i.e., heat coming from total of previously discharged assemblies). TVA has assumed the worst case combination of these two heat sources. The TVA analysis has quantified the actual TPC impact on core heat loads at approximately 0.3 MWt, which included both the decay heat generated by freshly discharged fuel assemblies during a refueling outage, and the additional residual decay heat from the increased discharge rate (96 per outage) of fuel assemblies into the pool. This value is based on conservative, full pool SFP conditions.

Increased Spent Fuel Pool Cooling Heat Rejection on ERCW

The design basis analysis for the ERCW was evaluated for impact from the increased heat load from the CCS. The increased SFPCCS heat load rejection to the CCS will not result in a significant temperature increase in ERCW. The higher proposed increase in allowable decay heat load in the SFP is comprised of both TPC related decay heat increase and additional margin to allow off loading fuel to the SFP as early as 100 hours. The increase in decay heat associated with TPC is approximately 1 MBTU/Hr. The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 10 MBTU/Hr. The proposed increase in decay heat above the approximate 1 MBTU/Hr associated with TPC, is decay heat that is shifted from the RHRS to the SFPCCS. The shifting results from the fact that fuel is either in the core being cooled by RHRS, or it is in the SFP being cooled by the SFPCCS. Since the decay heat has only shifted between systems, there is no net increase in CCS heat load on the ERCW system for this portion of the increased decay heat.

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

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

ERCW Summary

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

1.5.9 Ultimate Heat Sink

> *NUREG-1672, Section 2.9.1, "DOE evaluated the effect of TPBARs on the ultimate heat sink (UHS) for the reference plant against the guidance of SRP Section 9.2.5. The acceptance criteria specified in the SRP are based on meeting the relevant requirements of GDCs 2, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50. DOE states that the heat removal capability of the UHS may be affected by the TPC from the increase in the spent fuel pool heat load during cooldown operations and the subsequent effect on the component cooling water system and the station service water system. DOE concludes that the effect on the ultimate heat sink should be analyzed on a plant-specific basis. The staff agrees with this evaluation because the design of the ultimate heat sink is very plant-specific. The staff has identified this as an interface item that must be*

addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

NOTE: This item is being submitted separately.

The design basis function of the UHS is to provide an uninterrupted source of cooling water for decay heat removal. The maximum allowable inlet temperature for the UHS is 85°F. The ERCW System is utilized to supply water from the UHS to cool primarily safety related components. The CCS is the primary means for cooling the plant and removing residual decay heat during late stages of plant cooldown and during outages via its intermediate cooling loop providing a heat sink to the SFPCCS and RHR system.

Tritium Impact on Spent Fuel Pool Decay Heat

See previous discussion in Section 1.5.8.

Increased Spent Fuel Pool Cooling Heat Rejection on UHS

The design basis analysis for the UHS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant temperature increase in the UHS. The increase in decay heat associated with TPC is approximately 1 MBTU/Hr. The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 10 MBTU/Hr. This total increase in decay heat load is well within the design bases limiting heat load imposed on the ERCW and UHS during other modes of operation. Increased ERCW flows are the same higher flow rates that have been specified during other modes of operation. This small amount of increased decay heat and increased ERCW flow, when compared to the overall flow rates of the UHS through the ERCW System, produces an insignificant increase (< 0.1°F) in UHS temperature leaving the plant site. Since there is no significant increase, and since the ERCW has significant margin available, no changes to the ERCW temperature requirements are warranted.

UHS Summary

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

1.5.10 New and Spent Fuel Storage

NUREG-1672, Section 2.9.2, "The staff reviewed the effect of storing fuel assemblies with TPBAR assemblies in the new and spent fuel racks for the reference plant in accordance with SRP Section 9.1.1 for the new fuel storage and SRP Section 9.1.2 for the spent fuel storage. An analysis has previously been performed using the weight of 1470 pounds for a standard fuel assembly. The TPBARs, as burnable poisons, are similar in form to the Westinghouse standard *burnable poison rod assemblies (BPRAs). Because certain space on the storage racks for fuel assemblies will be replaced by TPBAR assemblies, the combined weight of a fuel assembly with TPBARs was calculated to be less than 1430 pounds. DOE also analyzed the dynamic effects for the TPBAR assembly that rests on the top nozzle adapter plate of the fuel assembly and found that the dynamic effect is insignificant. Because the weight of a fuel assembly with TPBARs is less than the weight of the standard fuel assembly previously analyzed, the staff concludes that the current design of the new and spent fuel pool facilities is still valid for the racks containing TPBAR assemblies. However, because the fuel rack analysis is plant-specific, the staff agrees with DOE's conclusion that the specific storage configuration for a plant participating in DOE's program for the CLWR production of tritium should be analyzed and could require changes to the TS. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

New Fuel Storage Vault

The current New Fuel Storage Vault criticality analysis has shown that unpoisoned fuel assemblies (without either discrete or integral poison) containing nominal enrichments up to 5.0 w/o ^{235}U can be stored in the fresh fuel rack array utilizing 120 specific cells of the 130 available storage locations. Fresh fuel containing TPBARs stored in the New Fuel Storage Vault will have a lower reactivity than unpoisoned fresh fuel assemblies. Therefore, the existing criticality analysis and New Fuel Storage Vault configuration remains conservative and valid when storing fuel assemblies containing TPBARs.

Spent Fuel Storage Pool

TVA has reanalyzed the criticality safety analysis for the installed (region 1) spent fuel storage racks. This reanalysis was performed with fuel assemblies of nominal enrichments up to 5.0 w/o ^{235}U containing TPBARs and also addressed other neutron poisons including Wet Annular Burnable Absorbers (WABA and Integral Fuel Burnable Absorbers (IFBA). The fuel was assumed to operate with TPBARs or WABAs, which were removed at the time the assemblies were placed in storage. Credit was taken for IFBA and fuel burnup, where appropriate.

The reanalysis demonstrated that sufficient conservatism was present in the previous analysis of the region 1 storage racks to adequately account for the effects of operating with TPBARs and confirmed that no Technical Specification changes were needed for these racks. However, the Technical Specifications also include limitations on non-installed fuel storage racks (region 2). These racks have not been reanalyzed to account for TPBAR effects. TVA has decided that the region 2 racks will not be utilized at Watts Bar and, therefore, will delete the region 2 related material from the Technical Specifications.

Analyses were also performed to determine the limiting amount of water that can be displaced in order to checkerboard non-fissile bearing components with fresh fuel. It was conservatively determined that 75% of water can be safely displaced in empty cells by non-fissile bearing components. Because a fully loaded TPBAR storage canister containing 300 TPBARs displaces approximately 51% of the water in a storage cell, no additional restrictions are necessary on the location of the TPBAR canister in the Spent Fuel Pool.

1.5.11 Spent Fuel Pool Cooling and Cleanup System

NUREG-1672, Section 2.9.3, "The staff has reviewed the information presented by DOE and concludes that the calculations performed by DOE may not represent the actual increase in pool temperature from incorporation of the TPBARs. However, on the basis of information submitted by DOE in its letter dated January 13, 1999, the decay heat generated by the TPBARs is very low; each TPBAR generates less than 3 watts of heat at 150 hours after reactor shutdown. The maximum temperature increase of a TPBAR due to internal heat generation is less than 3°F. The reference plant could insert up to 3344 TPBARs in each reload. The total heat load increase due to TPBARs is about 0.003 percent compared with a 3565 MWT core rating of the reference plant. In considering its very low rate of heat generation, the staff concludes that the heat load increase from the incorporation of TPBARs in the spent fuel pool has an insignificant impact on the spent fuel pool heat load and the added heat load will be within the cooling capability of the SFPCS. However, further analysis with reliable data is required to determine the actual impact of the TPBARs. A quantitative analysis to determine the absolute spent fuel pool temperatures must be performed by licensees seeking to utilize a TPC because the capacity of the spent fuel pool and its associated cooling system design are very plant specific. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

NOTE: This item is being submitted separately.

The SFPCCS for WBN is sized to handle full core off-loads. In the 1996-97 timeframe, WBN underwent spent fuel storage rack additions, which included development of a new thermal hydraulic analysis based on standard NRC approved methodologies which are scenario based. During the rerack design change TVA recognized the impracticality of following a scenario based set of limits during plant operation for predicting SFP decay heat load. During the licensing efforts associated with the rerack efforts at WBN, the FSAR was revised to capture a limiting value of decay heat that could be placed in the SFP, based on outage specific decay heat analysis performed for each outage. This approach provided a more realistic means (based on quantitative limits instead of a scenario based limits) of assuring compliance with the maximum allowable design basis decay heat loads that could be placed in the SFP at any time. Compliance with these limiting values provides assurance that, should a train of SFPCCS fail, maximum analyzed temperatures of the SFP and attendant decay heat removal system piping will not be exceeded.

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

Tritium Impact on SFP Decay Heat

See previous discussion in Section 1.5.8.

In addition, the impact of the higher heat load in the SFP could be mitigated by delaying the start of core off-load by 10 to 20 hours. Therefore from a design basis standpoint, it could be concluded that tritium production operations have no adverse impact on SFP heat loads or the ability of associated systems to remove the heat loads. However, since delaying the start of off-loading of the core during a plant outage results in a financial impact to plant operations, TVA has developed an alternate decay heat analysis which would compensate for this additional heat load and also accommodate core off-loading as early as 100 hours after shutdown.

Alternate SFP Decay Heat Analysis

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

SFP Heat Exchanger Fouling Factor

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

Component Cooling System Maximum Water Temperature

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

Results of Alternate Analysis

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

Table 1.5.11-1 WBN Spent Fuel Pool Design Parameters

Parameter	Existing Design Value	Proposed Value (Alternate Analysis)
Maximum Allowable Decay Heat Load	32.6 MBTU/Hr	32.6 - 47.4 MBTU/Hr See Note 1.
SFPCCS Flow	2300 GPM per Hx	2300 GPM per Hx
CCS Flow	3000 GPM per Hx	3000 GPM per Hx
Allowable Tube Plugging	5 %	5 %
Tube-Side Fouling (hr*ft^2*°F/BTU)	0.0005	0.0005 - 0.0001
Shell-Side Fouling (hr*ft^2*°F/BTU)	0.0005	0.0005 - 0.0001
Maximum CCS Temperature	95°F	95 - 80°F (Note 1)
Maximum SFP Temperature (1-Train)	159.24°F	159.24°F
Maximum SFP Temperature (2-Train)	129.30°F	129.30°F

Table 1.5.11-1 WBN Spent Fuel Pool Design Parameters (cont.)		
Parameter	**Existing Design Value**	**Proposed Value (Alternate Analysis)**
Average Time to SFP Boiling	5.24 Hours	3.4 Hours
Average SFP Heat-Up rate	10.2°F/Hr	15.54°F/Hr
Average Boil-Off Rate	70.20 GPM	102 GPM
Time until only 10 feet of water over racks - without makeup	43 Hours	29.8 Hours
Time until only 10 feet of water over racks - with 55 gpm makeup	200 Hours - Assuming constant decay heat. See Note 2	76 Hours
Margin to Localized Rack Boiling	9.6°F	6.8°F

Notes:

1. The range of values represent allowable heat loads based on specific combinations of heat exchanger fouling between 0.0005 and 0.0001 ($hr*ft^2*°F/BTU$) and actual CCS temperatures between 95 to 80°F.
2. If credit is taken for decreasing core decay heat energy during the 200 hour period, the 10 feet above rack level is never reached at a makeup rate of 55 GPM.

Impact of Higher Allowable Decay Heat in the SFP

As shown in the table above, the proposed change will not result in an increase in maximum SFP temperature. The only operational effect is noted during complete loss of both trains of cooling, whereby the higher allowable decay heat results in higher boil-off rates and faster required response times to mitigate the loss of SFP cooling event. The proposed values above, however, are comparable to existing values at Sequoyah Nuclear Plant, which was licensed for a higher allowable decay heat load during its rerack project.

An analysis has also been performed to evaluate the effect on localized temperatures within a spent fuel rack. The analysis was performed consistent with existing analysis methodologies except the rack and pool area were modeled using a three dimensional nodalization, instead of two dimensional. The inputs were revised to be consistent with the maximum allowable decay heat value (47.4 MBTU/hr). The results of the analysis show that while the margin to localized boiling has decreased, localized boiling within a rack will not occur. The analysis specifically concluded that:

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

2. the bounding local fuel cladding temperature in the racks, determined by adding the bounding temperature difference between cladding material and water in the racks to the maximum local water temperature, was less than the local saturation temperature.

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

While existing design limits and operational procedures are adequate to prevent exceeding design limits on allowable SFP heat load, TVA proposes to revise the allowable heat loads. TVA proposes to increase the maximum allowable decay heat in the WBN SFP from 32.6 MBTU/Hr to a range between 32.6 MBTU/Hr and 47.4 MBTU/Hr. The lower value of 32.6 MBTU/Hr will only be exceeded if actual operating conditions of lower CCS temperature and/or lower than design fouling is present. Specific curves relating CCS Temperature and SFP heat exchanger fouling to allowable SFP decay heat have been developed to assist Operations in evaluating allowable SFP decay heat for each core off-loading evolution. These higher values of allowable decay heat within the SFP will not result in exceeding the analyzed maximum SFP temperature under normal full core off-load conditions (two train operation) of 129.3°F, and a faulted maximum temperature (one train operation) of 159.2°F. TVA is seeking a licensing change to its SFPCCS allowable heat loads to allow use of actual fouling factors and CCS temperature in lieu of design values.

SFPCCS Summary

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

1.5.12 Component Cooling Water System

NUREG-1672, Section 2.9.4, "Because more fuel and TPBAR assemblies are removed from the core to the spent fuel pool during refueling, the maximum pool temperature will increase. Although the effect of the TPBARs on the CCWS is insignificant because the heat load generated by the TPBARs only amounts to about 3 watts per rod 150 hours after reactor shutdown, a substantial increase in heat load occurs as a result of a full core off-load. The additional heat load generated by the TPC to the spent fuel pool heat exchangers could increase the demand for CCWS flow. DOE stated that the system heat transfer and flow requirements may be affected by the TPBARs from the increase in spent fuel pool heat load during cooldown operations, and the effect on this system will need to be analyzed on a plant-specific basis. In response to the staff's RAI, DOE also stated that the increased spent fuel pool heat load does not come from the presence of TPBARs but from the increased number of fuel assemblies being replaced. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

NOTE: This item is being submitted separately.

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

Tritium Impact on Spent Fuel Pool Decay Heat

TVA has prepared a quantitative analysis of expected spent fuel decay heat for both Tritium Production Cores (TPCs) and non-TPCs. The analysis is based on comparative decay heat data prepared by TVA for a base non-tritium core, a TPC with 80 fresh fuel assembles (80-feed), and a TPC with 96 fresh fuel assemblies (96-feed). The results of the analysis show that the 80 feed case was limiting for decay heat, and the 80-feed TPC core contributes a slightly higher decay heat over the non-TPC and the 96-feed TPC, due to isotopic composition differences between the base and TPC cores, for the same design basis reactor power level. The results of the analysis show that the 96-feed case was limiting for residual heat. The TVA analysis has quantified the actual TPC impact on core heat loads at approximately 0.3 MWt (approximately 1 MBTU/HR), which included both the decay heat generated by freshly discharged fuel assemblies during a refueling outage, and the additional residual decay heat from the increased discharge rate (96 per outage) of fuel assemblies into the pool. This value is based on conservative, full pool SFP conditions.

Increased Spent Fuel Pool Cooling Heat Rejection on CCS

The design basis analysis for the CCS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant temperature increase on CCS. The increase in decay heat associated with TPC is approximately 1 MBTU/Hr. This decay heat load increase is approximately 1% of the total design heat load on the CCS. The higher proposed increase in allowable decay heat load in the SFP, however, is comprised of both TPC related decay heat increase, plus additional margin to allow commencement of core off loading activities as early as 100 hours after shutdown. The proposed increase in decay heat above the approximate 1 MBTU/Hr associated with TPC, is a CCS heat load that is shifted from the RHRS to the SFPCCS. The shifting results from the fact that fuel is either in the core being cooled by RHRS, or it is in the SFP being cooled by the SFPCCS, both systems ultimately rejecting their respective heat burdens on the CCS.

CCS design thermal analyses have been evaluated and determined to be capable of accepting the increased SFPCCS allowable decay heat loads. CCS flows to the SFPCCS heat exchangers have not been increased. The additional heat load rejected to the CCS from the SFPCCS heat exchanger results in slightly elevated CCS temperatures, but is within existing design basis values. Piping analysis and support analysis of the CCS have been previously analyzed at a higher ultimate temperature associated with more bounding operational modes, and are not affected by the increased CCS heat load. The downstream dilution effect also helps to minimize the impact of the elevated CCS temperatures, since as SFPCCS heat loads increase, the RHRS heat loads decrease. With all CCS flows returning to a common header prior to returning to the CCS/ERCW heat exchangers, there is no measurable change to the mixed stream CCS temperature.

Since higher allowable SFP decay heat can be placed in the SFP if CCS temperatures and/or SFP heat exchanger fouling factors are shown to be less than design, maintaining the CCS temperature during outages to as low as possible is desired. CCS temperatures can be lowered considerably if ERCW flows to the CCS heat exchangers are increased. Increased ERCW flow rates are within existing flow criteria established for other modes of operations.

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

CCS Summary

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

1.5.13 Demineralized Water Makeup System

NUREG-1672, Section 2.9.5, "The staff has reviewed the information presented by DOE and concludes that the incorporation of TPBARs in the reference plant does not have any significant impact on the demineralized water makeup system because only a very small quantity of tritium is released from the TPBARs to the primary coolant system. Because the design of the demineralized water makeup system is plant-specific, DOE concludes, and the staff agrees, that a detailed analysis for this effect is required from licensees participating in DOE's program for the CLWR production of tritium. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

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

TVA does not intend changes to the plant's current feed and bleed operations to control boron concentration in the RCS. Continuation of the current feed and bleed program will result in the RCS observed maximum tritium levels of 2.5 µCi/gm increasing to around 9 µCi/gm with the TPC. This increase is due to normal reactor tritium production plus the tritium permeation from TPBARs. Public doses from liquid and airborne effluent release will remain below applicable ODCM limits, and tritium release concentrations will remain within 10 CFR 20 and ODCM release limits.

In the abnormal event of two TPBAR failures, RCS tritium values could increase to approximately 105 μCi/gm. Following this unlikely event, approximately 150,000 gallons of additional feed and bleed would be necessary to reduce the tritium concentration to the 9 μCi/gm range. This estimate is based on the failures occurring near the end of the cycle. However, public doses from liquid and airborne effluent release will remain below applicable ODCM limits, and tritium release concentrations will remain within 10 CFR 20 and ODCM release limits.

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

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

TVA's review of the DWMS for WBN has determined that the current system's storage and water production capacity, compared to the expected increase in feed and bleed required to mitigate a two TPBAR failure event, is adequate. Public doses from liquid and airborne effluent release will remain below applicable ODCM limits, and tritium release concentrations will remain within 10 CFR 20 and ODCM release limits. See Sections 1.5.14 and 2.11.3 for more information concerning Liquid Waste Management.

The DWMS and storage tanks will not require modification, nor will the water supply contract require changes to support tritium production activities at WBN.

1.5.14 Liquid Waste Management Systems

> *NUREG-1672, Section 2.11.2, "On the basis of the preceding discussion, the staff concludes that in both cases (the design-basis TPBAR permeation of tritium and the failure of two TPBARs) there is a sufficient margin in the reference plant so that the applicable release concentration and dose limits as presented in the plant technical specifications and ODCM will still be met even with the TPC operation. However, enhanced plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event are required. Furthermore, when the TPC topical report is applied to a candidate plant, a plant-specific analysis will be needed to demonstrate that the plant continuously meets release concentration and dose limits. The staff concludes that the methodology described in Section 2.11.3 of the TPC topical report is acceptable for plant-specific analysis. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."*

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

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

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

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

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

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

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

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

However, doses from liquid and airborne effluent release will remain below applicable ODCM limits, and tritium release concentrations will remain within 10 CFR 20 and ODCM release limits. For additional information refer to Section 2.11.3.

1.5.15 Process and Effluent Radiological Monitoring and Sampling System

NUREG-1672, Section 2.11.5, "In Section 2.11.6 of the TPC topical report, DOE states that the current process and effluent radiological monitoring instrumentation and sampling systems that are in place at the reference plant, as well as at other operating PWR plants, include the capability for monitoring the tritium levels within the plant and in plant effluent pathways, and are adequate for use when the plant is operated with a TPC. On the basis of its review, the staff agrees with DOE that the existing capability for radiation monitoring is adequate for tritium levels at the reference plant. In response to the staff's RAI dated October 15, 1998, DOE stated that the details of the laboratory instrumentation and sampling frequencies and locations are plant dependent. Therefore, a plant-specific assessment of the candidate plant for the TPC will be required to provide such information. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

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

Tritium Monitoring

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

Air Sampling

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

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

Liquid Monitoring

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

Liquid Scintillation Counting

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

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

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

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

1.5.16 Use of LOCTA_JR Code for LOCA Analyses

> *NUREG-1672, Section 2.15.5, "The staff concludes from its review that calculated TPBAR performance under LOCA conditions has demonstrated that TPBARs can be assessed with approved licensing LOCA models and can perform acceptably under LOCA conditions. However, the staff also concludes that, although the LOCTAJR code was appropriate for use in the demonstration analyses and assessments discussed herein, LOCTAJR was not reviewed for licensing use and should be reviewed by the staff for licensing applications and for its interface with the specific plant licensing LOCA models before it is used in specific plant licensing applications."*

In the references listed below, TVA provided the required information concerning use of the LOCTA_JR code. Subsequently, NRC issued a safety evaluation dated January 17, 2001, documenting its acceptance of this code for use in licensing analyses and closing this interface item.

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

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

1.5.17 ATWS Analysis

NUREG-1672, Section 2.15.7, "The staff agrees with the partial ATWS analysis conducted and the results obtained by DOE. However, this concurrence pertains only to the TPC topical report. The staff concludes that licensees seeking to utilize a TPC must submit a plant-specific application containing a full ATWS analysis, conducted in accordance with NRC regulations and approved standards. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

In the reference listed below, TVA has submitted the ATWS analysis for NRC staff review. Subsequently, NRC issued a safety evaluation dated March 16, 2001, documenting its acceptance of the TVA response.

- Letter from TVA (P. L. Pace) to NRC Document Control Desk dated September 29, 2000, regarding WATTS BAR NUCLEAR PLANTS (WBN) – TRITIUM PRODUCTION – ANTICIPATED TRANSIENTS WITHOUT SCRAMS (ATWS).

1.6 WATTS BAR PLANT SPECIFIC CHANGES

During the NRC's review of Reference 1, the NRC determined that a facility undertaking irradiation of a tritium production core will require changes to the Technical Specifications (TS) contained in Appendix A of their facility operating license. The evaluations and analyses for Watts Bar contained in this report along with References 1 and 3 provide the technical bases for the Watts Bar TS changes necessary to irradiate TPBARs.

1.6.1 Technical Specifications

The following TS sections were identified in the SER as candidates for change.

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

1.6.2 Watts Bar Specific TS Changes

TVA has evaluated the use of TPBARs in Watts Bar Unit 1 and has determined that the following TS sections require modification to support TPBAR implementation:

1. TS 3.5.1 - Cold Leg Accumulator – Boron Concentration Increase
2. TS 3.5.5 - Refueling Water Storage Tank – Boron Concentration Increase
3. TS 4.2.1 - Design Features, Fuel Assemblies

TVA requests these TS changes via this amendment to the Watts Bar operating license to allow operation with a tritium production core.

The NRC in Reference 3 identified several potential TS changes (see Section 1.6.1) which could be required to support operation with TPBARs. The identified TS changes and their applicability to Watts Bar are discussed below:

- TS 3.4.3 – RCS Pressure and Temperature (P/T) Limits

  The Watts Bar Technical Specifications do not contain pressure/temperature (P/T) limit curves, but instead reference the Pressure Temperature Limits Report (PTLR), therefore no Technical Specification change would be required if the curves were revised. Furthermore, it has been demonstrated (see Section 1.5.4) that placing TPBARs in specific peripheral assemblies suppresses the power in those assemblies and maintains the vessel fluence to a level comparable to current low leakage core designs within normal cycle to cycle variability. Therefore, there is no change to the Appendix G P/T limit curves in the PTLR.

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

  The Watts Bar Technical Specifications do not contain the LTOP setpoints, but instead reference the Pressure Temperature Limits Report (PTLR), therefore no Technical Specification change would be required if the setpoints were revised. Furthermore, it has been demonstrated that the existing Appendix G limit curves in the PTLR remain applicable (see Section 1.5.4) and, consequently, the existing LTOPS analyses and setpoints remain applicable for the Tritium Production Core. Therefore, there is no change to the LTOPS setpoints in the PTLR.

- TS 3.7.17 (WBN TS - 3.7.15) - Spent Fuel Assembly Storage
  TS 4.3 - Design Features, Fuel Storage

These Technical Specifications denote criticality requirements related to fuel storage in the New Fuel Storage Vault and Spent Fuel Storage Pool storage racks. The criticality effects of TPBARs on these storage racks have been discussed in Section 1.5.10. The existing New Fuel Storage Vault rack criticality analysis remains conservative and valid when storing fuel assemblies containing TPBARs and, therefore, no Technical Specifications changes are required for fuel assembly storage in the New Fuel Storage Vault. A reanalysis of the currently installed (region 1) spent fuel storage racks has demonstrated sufficient conservatism to adequately accommodate the effects of operating with TPBARs and confirmed that no Technical Specifications changes were required for these racks. However, the Technical Specifications also include limitations on non-installed fuel storage racks (region 2). These racks have not been reanalyzed to account for TPBAR effects. TVA has decided that the region 2 racks will not be utilized at Watts Bar and, therefore, will delete the region 2 material from the Technical Specifications.

1.7 REFERENCES

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

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

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

4. Letters from TVA to NRC, dated June 23, 2000 and October 5, 2000.

5. Letter from TVA to NRC, dated September 29, 2000.

6. WCAP-14565-P-A, "VIPRE-01 Modeling Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

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

8. Emergency Response Guidelines - Revision 1B, Westinghouse Owners Group, 2/28/92.

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

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