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

September 21, 2001

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

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

Gentlemen:

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

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - REVISION OF INSTRUMENTATION MEASUREMENT RANGE, BORON CONCENTRATION LIMITS, REACTOR CORE LIMITATIONS, AND SPENT FUEL POOL STORAGE REQUIREMENTS FOR TRITIUM PRODUCTION CORES (TPCs) - TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-06

In accordance with the provisions of 10 CFR 50.90, TVA is submitting a request for an amendment to SQN's Licenses DPR-77 and DPR-79 to change the TSs for Units 1 and 2 to allow SQN to provide irradiation services for the U.S. Department of Energy (DOE). This change would allow SQN 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 revising the measurement range for the source range monitors in TS Table 3.3-9, increasing the required boron concentration for both the cold leg accumulators (TS 3/4.5.1) and the refueling water storage tank (TS 3/4.5.5), deleting the boron concentration and spent fuel storage requirements and associated Bases for the cask pit pool in TS Section 3/4.7.14 and Section 5.6, adding a limit on the number of TPBARs that can be irradiated in TS Section 5.3, providing storage requirements for spent fuel assemblies that contained TPBARs during irradiation in TS Section 5.6 and the Bases for TS Section 3/4.7.13, and the implementation of a TPBAR consolidation activity. This submittal also provides revisions to the TS Bases in Section 3/4.6.4 associated with combustible gas control.

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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 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 loss-of-coolant accident (LOCA) material ejection from the TPBARs. Both issues incorporate current research and have been factored into the enclosed safety analyses. TVA has requested that DOE perform additional testing and analysis as described in Enclosure 4.

TVA has determined that there are no significant hazards considerations associated with the proposed change. The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed these proposed changes and have determined that operation of SQN Units 1 and 2, in accordance with the 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 to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed TS 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.

Enclosure 4 provides Framatome-Advanced Nuclear Power (ANP) Report BAW-10237, Revision 1 which:

- contains information relative to items in the TPC Topical Report for which there is a SQN 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 NUREG-1672 such as the TPBAR surveillance program, lead test assembly (LTA) post irradiation results, and a discussion of proposed TS changes identified in NUREG-1672 that are not required at SQN.

Although the SQN thermal power uprate of 1.3 percent is not required for the implementation and utilization of TPBARs, TVA anticipates, subsequent to NRC approval, the implementation of a thermal power uprate prior to initial insertion of the TPBARS into SQN Unit 1 or 2. Accordingly, those evaluations and analyses contained in the Framatome-ANP Report have enveloped the uprated power level of 3455 megawatt thermal (MWt) versus the current rating of 3411 MWt.

Portions of Enclosures 1 (TPBAR consolidation activity) and Enclosure 4 were previously submitted on May 25, 2001. In that submittal, areas labeled as "Information to be provided later," were identified. This submittal provides that information. The May 25th submittal also provided information regarding a new methodology for the spent fuel pool cooling analysis. TVA's Watts Bar Nuclear Plant (WBN) has requested NRC review and approval for this methodology change, in accordance with 10 CFR 50.90 in a submittal to NRC dated April 20, 2001. NRC's approval of this effort is expected to be completed before the date that the new methodology will be needed for SQN. Since both TVA sites will use the new methodology in the same manner, SQN will be able to implement this change in accordance with the 10 CFR 50.59 requirements after NRC's approval of the WBN request.

Therefore, this submittal does not include a duplicate request for NRC review.

In order to meet DOE's Tritium Program requirements, TVA requests that this amendment be approved within one year of this submittal date and that the revised TSs be made effective during each unit's respective Cycle 12 refueling outage in order to properly implement the boron concentration changes.

There are no new regulatory commitments being made by this submittal. This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this license amendment request, please contact Pedro Salas at (423) 843-7170.

Sincerely,

Dennis L Koehl

Dennis L. Koehl
Plant Manager

Enclosures

Subscribed and sworn to before me
on this 21th day of September

Peggy M Billingsley
Notary Public

My Commission Expires October 9, 2002

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EDMS, WTC A-K

i:License/TS Submittal/TS Change 00-06

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NO. 327, 328

**PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS 00-06
AND TPBAR CONSOLIDATION ACTIVITY
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE**

PART A - PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

I. DESCRIPTION OF THE PROPOSED CHANGE

In order to irradiate Tritium Producing Burnable Absorber Rods (TPBARs) at SQN, changes to six sections of the TSs, along with the appropriate Bases discussions and one TS Bases discussion, need to be made. The first change revises the measurement range for the backup source range monitor. The next two changes involve increasing the boron concentration in both Cold Leg Accumulators (CLAs) and Refueling Water Storage Tank (RWST) which stem from fuel core design. The fourth change deletes the provisions for storing spent fuel in the cask pit and the associated boron concentration requirements. The fifth change involves incorporating into the Design Features Section 5.0 the maximum number of TPBARs that can be inserted into the reactor core in an operating cycle. The sixth change adds discussions regarding fuel assemblies that contained TPBARs during a fuel cycle and the applicable storage requirements. A revision to the TS Bases discussion for combustible gas control has also been included to properly describe the possible sources of hydrogen gas. Each of these changes are described below and illustrated in Enclosures 2 and 3:

A. TS Table 3.3-9 - Remote Shutdown Monitoring Instrumentation - Revised Backup Source Range Monitor Measurement Range

This change will revise the measurement range of the backup source range monitor. The current range is from 1 to 10^6 counts per second (CPS) and the proposed range is from 0.1 to 10^5 CPS.

B. TS 3/4.5.1 - Cold Leg Injection Accumulators - Boron Concentration Increase

This change is requested to increase the CLA Boron Concentration from the present range of 2400 to 2700 parts per million (ppm) to a range of 3500 to 3800 ppm.

C. TS 3/4.5.5 - Refueling Water Storage Tank - Boron Concentration Increase

This change is requested to increase the RWST Boron Concentration from the present range of 2500 to 2700 ppm to a range of 3600 to 3800 ppm.

D. TS 3/4.7.14 and Bases - Cask Pit Pool Minimum Boron Concentration - Deletion of Requirements

This TS section and the associated Bases discussions are being deleted in their entirety.

E. TS 5.3.1 - Design Features/Reactor Core/Fuel Assemblies

A change is requested to Section 5.0, Design Features, to allow the insertion of a maximum of 2256 TPBARs into the SQN 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 2256 TPBARs.

This request would insert a new sentence to Section 5.3.1 to read as follows:

> Sequoyah is authorized to place a maximum of 2256 Tritium Producing Burnable Absorber Rods into the reactor core in an operating cycle.

F. TS 5.6 and TS 3/4.7.13 Bases - Design Features/Fuel Storage and Spent Fuel Pool Minimum Boron Concentration - Revised Storage Requirements for Fuel Assemblies Containing TPBARs

Current Section 5.6.1.1 is being revised to accommodate new provisions that address the storage of spent fuel that contained TPBARs. Information has been included at the beginning of this section to define Type A fuel (spent fuel that has not contained TPBARs), Type T fuel (spent fuel that has contained TPBARs), fresh fuel, and cooling time. Spent fuel pool Region 1 is designated to contain fresh fuel and spent fuel Type A. Region 2 is designed to contain spent fuel Type A or Type T. Region 3 is designated to contain fresh fuel only. Region 4 is designated to contain fresh fuel and spent fuel Type T. As part of the revisions to Section 5.6.1.1, clarifying information regarding storage cells partially filled with non-fissile material has been included for all regions. This revision also deletes current Section

5.6.1.1.d that addresses spent fuel storage provisions for the cask pit pool.

Section 5.6.3 is also revised to delete the last sentence that reads:

> In addition, no more that 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

The figures and tables associated with Section 5.6.1.1 have been revised accordingly to properly represent the acceptable spent fuel storage patterns for each fuel type with appropriate enrichment, burnup, and cooling time requirements for storage in respective regions of the spent fuel pit. This change has revised the labels for the existing figures and tables to clarify their use for Type A spent fuel and has included other changes to reflect the new Type T spent fuel. New figures and tables for Type T spent fuel have been added with appropriate labels and information for controlling storage requirements for this fuel.

Additionally, the Bases for spent fuel pool boron concentration for TS Section 3/4.7.13 has been revised to be consistent with the changes to TS Section 5.6.1.1. These changes reflect the use of TPBARs in fuel assemblies, the storage of Type A and Type T spent fuel, the designations for Regions 1 through 4 of the spent fuel pool, and the associated reference additions.

G. Bases 3/4.6.4 - Combustible Gas Control - Hydrogen Generation Sources

As a result of the Tritium Program, a change is being made to the TS Bases for combustible gas control to include the hydrogen and tritium inside the TPBARs as possible sources. This change would insert a fourth hydrogen generation item into the discussions as follows:

> , and 4) tritium and hydrogen that exist in the Tritium Producing Burnable Absorber Rods prior to the accident.

II. REASON FOR THE PROPOSED CHANGE

A. TS Table 3.3-9 - Remote Shutdown Monitoring Instrumentation - Revised Backup Source Range Monitor Measurement Range

The current measurement range for the backup source range monitor provides an acceptable range of values for the current fuel loading configurations and the typical boration levels of the reactor coolant system (RCS). With the higher levels of boron concentrations that will be utilized with the tritium production cores (TPCs), the availability of neutrons to be detected by the backup source range monitor will be reduced. Therefore, lowering the measurement range of the monitor by one decade will provide a more adequate range that will bound the amount of neutrons that will be available for detection. This will result in indications within a more accurate portion of the monitor's indication capabilities.

B. TS 3/4.5.1 - Cold Leg Injection Accumulators - Boron Concentration Increase

The post loss-of-coolant accident (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 and without taking credit for any rod cluster control assembly insertion. 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 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 2000 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.B.

C. TS 3/4.5.5 - Refueling Water Storage Tank Boron Concentration Increase

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

D. TS 3/4.7.14 and Bases - Cask Pit Pool Minimum Boron Concentration - Deletion of Requirements

TVA requested the inclusion of TS Section 3/4.7.14 and received approval in SQN Amendment Nos. 265 and 256 for Units 1 and 2, respectively. These amendments provided for the storage of spent fuel in the cask pit pool in the event that additional room might be needed. TVA now intends to utilize the dry cask storage provisions for additional storage space. This, combined with the need to use the cask pit pool for TPBAR consolidation and dry cask storage activities, has eliminated the need to use this area for spent fuel storage. Therefore, TVA no longer plans to use this area for spent fuel storage and the provisions that allowed this use, as well as the boron concentration requirements, are being deleted in the proposed request.

E. TS 5.3.1 - 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 never will be greater than 2256 TPBARs.

F. TS 5.6 and TS 3/4.7.13 Bases - Design Features/Fuel Storage and Spent Fuel Pool Minimum Boron Concentration - Revised Storage Requirements for Fuel Assemblies Containing TPBARs.

TVA will be producing tritium in TPBARs as part of an agreement with the Department of Energy (DOE). As a result, spent fuel assemblies associated with the tritium production will require storage in the spent fuel pool and will have different characteristics than other non-tritium producing spent fuel. The TPBAR related fuel will be more reactive than other fuel and therefore will require more restrictive storage limitations. The proposed changes for TS Section 5.6.1.1 and the Bases for TS Section 3/4.7.13 for Type A and Type T spent fuel will provide appropriate requirements to ensure acceptable storage arrangements that will maintain the necessary criticality limits.

The change to current TS Sections 5.6.1.1.d and 5.6.3 eliminates the provision to store spent fuel in the cask pit pool consistent with the proposed deletion of TS Section 3/4.7.14 described above in Section II.D.

G. Bases 3/4.6.4 - Combustible Gas Control - Hydrogen Generation Sources

The purpose for the addition of a fourth hydrogen source for the combustible gas control 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 Table 3.3-9 - Remote Shutdown Monitoring Instrumentation - Revised Backup Source Range Monitor Measurement Range

The backup source range monitor provides an indication of core criticality conditions in the auxiliary control room. This monitor would be used in the event the main control room was required to be evacuated and shutdown conditions had to be monitored in a remote location. This monitor is used for indication of the core shutdown conditions and does not include the trip functions associated with the main control room monitors that support plant startup functions.

With the higher levels of boron concentrations that will be utilized with the TPCs, the availability of neutrons to be detected by the backup source range monitor will be reduced. Therefore, lowering the measurement range of the monitor by one decade will provide a more adequate range that will bound the amount of neutrons that will be available for detection during shutdown conditions. This change improves the ability to monitor neutron activity for verification of shutdown conditions which is the primary function of the monitor. This monitor has no startup or trip functions and therefore, there is no adverse impact for startup or operating conditions since these evolutions are handled by the main control room source range monitors.

While the bottom end of the monitor's range is lower, likewise the top end is also lowered by one decade thereby preserving the existing overall loop accuracy. The range of neutron activity during shutdown conditions will not be of a magnitude that the reduction of the upper end of the range will affect

the ability to verify shutdown conditions. The monitor will have equivalent or better capabilities to monitor changes in neutron activity with the revised measurement range to support the verification of unit shutdown. Since the backup source range monitors are used for indication of unit shutdown conditions and the lowering of the measurement range serves to improve this ability for lower leakage tritium cores, the proposed change is acceptable and no adverse impact to nuclear safety will result.

B. TS 3/4.5.1 - Cold Leg Injection Accumulators - Boron Concentration Increase

1. LOCA Related Analyses

a. Large Break LOCA (LBLOCA)

During an LBLOCA, the core becomes subcritical due to voids generated by the rapid system depressurization. Any additional boron injected due to the increase in the concentration levels would increase the margin by which the core is maintained in a subcritical condition. The LBLOCA analysis, however, does not explicitly model the boron concentration level of the accumulators or RWST; the calculated Peak Clad Temperature (PCT) and clad oxidation is not a function of the boron concentration. Thus, an increase in the accumulator and RWST boron concentrations would have no adverse effect on the 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. Though not modeled in the analysis, any additional boron injected due to the increase in the concentration levels would increase the margin by which the core is maintained in a subcritical condition. The calculated PCT and clad oxidation 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 emergency core cooling system (ECCS) water residing in the sump following a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water will result in the reactor core remaining subcritical assuming all control rods are out. 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.

Calculations have been performed to confirm that the sump solution will contain adequate boron to maintain the reactor in a shutdown condition following a LOCA. These calculations demonstrate that the required boron concentration to maintain subcriticality for the evaluated TPC is well below the mixed mean sump concentration. Reload TPCs will be evaluated to ensure continued compliance with this shutdown requirement.

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 long-term cooling problems associated with boron precipitation in the reactor vessel and core. The switchover 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 reduce the plate out of boron, 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 sump boron concentration is higher, and the threshold for boron precipitation and possible core coolant blockage occurs sooner. In order to assure the same degree of long-term cooling with the higher boron concentration, the current hot leg switchover value of 12 hours will be reduced to 5.5 hours. TVA has determined that the shorter hot leg switchover time does not impose an adverse burden on plant operators.

2. Non-LOCA Transient Analysis

The following non-LOCA accidents model the RWST boron concentrations and the accumulators do not inject.

a. Steamline Break (SLB) at Hot Zero Power

Following a SLB, a safety injection (SI) signal occurs as a result of low steam generator pressure and the ECCS provides borated water from the RWST to the RCS. An increase in RWST boron concentration could be expected to reduce post-break core power. For the worst-case SLB, however, dry-out of the broken steam generator and a subsequent reduction in RCS cooling ends the core power excursion prior to the introduction of boron into the RCS. The core power excursion is, therefore, not sensitive to boron addition. Therefore, an increase in boron concentration in the RWST and accumulators has no effect on the SLB analyses.

b. Feedwater Line Break

Following a feedwater line break, a SI signal can occur as a result of low steam generator pressure and the ECCS provides borated water from the RWST to the RCS. A reactor trip occurs and an increase in RWST boron concentration could be considered as additional shutdown reactivity added to the core. However, no credit for boration is conservatively taken in the analysis. The increase in RWST and accumulator boron concentration required by the TPBAR core design, therefore, has no effect on

the feedwater line break analyses.

c. Spurious Operation of the SI System at Power

This event is initiated by SI actuation. A spurious SI event is postulated to maximize the insertion of negative reactivity and assumes a maximum boron concentration. At the time the Sequoyah Final Safety Analysis Report (FSAR) analysis was performed, the boron injection tank (BIT) contained water borated to a concentration of 20,000 ppm. After the BIT concentration was reduced, the analysis was not revised as the high boron concentration was conservative. Because such a high boron concentration is considered in this event, an increase in the RWST boron concentration to as much as 3800 ppm is bounded by the current analysis. An increase in the RWST and accumulator boron concentration, therefore, does not affect the analysis of a spurious SI event.

3. 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.

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 SGTR analysis does not model the boron in the accumulators or the RWST. Therefore, there is no impact on the analysis.

5. Containment M&E 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 degrees Fahrenheit (°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 Sequoyah Units 1 and 2 RWSTs 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.

In the Westinghouse EQ program, documented as WCAP-8587, 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, however, it will remain within the current SQN lower limit of 7.5 pH. The pH reduction 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.

C. TS 3/4.5.5 - Refueling Water Storage Tank - Boron Concentration Increase

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

D. TS 3/4.7.14 and Bases - Cask Pit Pool Minimum Boron Concentration - Deletion of Requirements

TVA has not stored spent fuel in the cask pit and does not have plans to in the future. Since this TS requirement only addresses the potential for storage of spent fuel in the cask pit pool, the elimination will

not have any adverse impact since the storage function was never utilized and a specific boron concentration is not required. If TVA chooses to utilize this area for spent fuel storage in the future, the appropriate analysis, along with a license amendment request to NRC, will have to be processed. Elimination of this requirement, along with the deletion of other provisions to allow storage in the cask pit pool, will not impact nuclear safety. Boron concentration will continue to be properly maintained for the storage of spent fuel in the spent fuel pool to control inadvertent criticality events.

E. TS 5.3.1 - Design Features/Reactor Core/Fuel Assemblies

The proposed change is justified based on extensive analysis, testing, and evaluation of the TPBARs as reported previously in the TPC Topical Report and on the evaluations performed for SQN described in Framatome-Advanced Nuclear Power (ANP) Topical Report BAW-10237. TVA has performed the confirming checks recommended by the DOE TPC Topical Report and plant specific evaluations requested by NRC's 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.

F. TS 5.6 and TS 3/4.7.13 Bases - Design Features/Fuel Storage and Spent Fuel Pool Minimum Boron Concentration - Revised Storage Requirements for Fuel Assemblies Containing TPBARs

For spent fuel pool storage, fuel is divided into three categories: spent fuel that has hosted TPBARs (designated Type T fuel), spent fuel that has not hosted TPBARs (designated Type A fuel), and fresh fuel. Fresh fuel can be stored in Regions 1, 3, or 4. Type A spent fuel can be stored in Regions 1 or 2 if the appropriate enrichment, burnup, and cooling time thresholds are met. Type T spent fuel can be stored in Regions 2 or 4 if the appropriate burnup and cooling time thresholds are met.

Design Feature 5.6.1.1 requirements pertaining to Type A spent fuel are unchanged from the current Design Feature 5.6.1.1 except for: (1) the clarification that storage of miscellaneous items or equipment displacing no more than 75% of cell volume applies to all regions and (2) the deletion of the 15 x 15 cask loading pit storage rack since this option will not be used. (The cask pit rack is also deleted from Design Feature 5.6.3). The previous criticality safety analysis (Holtec International Report HI-992349, Rev. 1) and boron dilution analysis (Holtec International Report HI-992302, Rev. 1) supporting TS Change 99-17 (Soluble Boron Credit) still apply to, and fully support, storage of Type A spent fuel.

Design Feature 5.6.1.1 requirements pertaining to Type T spent fuel are structured similar to the requirements for Type A spent fuel. A new storage region (Region 4) is defined for fresh fuel and Type T spent fuel in the same 1-of-4 pattern as Region 1 has for fresh fuel and Type A spent fuel but with different burnup and cooling time thresholds for the Type T spent fuel. Region 2 storage can intermingle Type A and Type T fuel but with separate enrichment, burnup and cooling time thresholds for each type fuel.

Region 3 is designed to store fresh fuel in a 2 of 4 array of fresh fuel assemblies and water filled cells. The presence or non-presence of TPBARs is immaterial for fresh fuel.

The criticality safety analysis for the spent fuel storage racks has been reanalyzed (Holtec International Report HI-2012629). This reanalysis was performed with fuel assemblies of nominal enrichments of 5.0 weight percent U235 containing TPBARs (Type T fuel) and also addressed other neutron poisons including Burnable Poison Rod Assemblies (BPRAs) and Gadolinia integral absorber rods (Type A fuel). The fuel was assumed to operate in-core with TPBARs, which were removed at the time the assemblies were placed in the spent fuel pool. As in the current analysis, credit was taken for soluble boron, fuel burnup, and cooling times, where appropriate.

The reanalysis adequately accounted for the effects of operating with TPBARs and determined burnup versus cooling time curves applicable to fuel burned with TPBARs for the various storage regions. The composition of the storage regions (i.e., 1 of 4 checkerboard, 2 of 4 checkerboard, or solid matrix) remains the same as in current TSs, but with different burnup and cooling time thresholds and with Regions 1 and 4 being limited to Types A and T spent fuel, respectively. The results of the reanalysis assure a

safe storage configuration of fresh and spent fuel assemblies in the spent fuel pool.

G. Bases 3/4.6.4 - Combustible Gas Control - Hydrogen Generation Sources

The addition of a new source for hydrogen gas in the Bases only serves to completely describe considerations included in the evaluation for TPBAR irradiation. These changes do not alter the TS requirements or the functions for the combustible gas control features at SQN. 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 NRC accepted Quality Assurance Program. It will normally be stored in the cask lay-down area when not in use. The TCF includes a video monitoring system, lighting, and tools designed to remove TPBARs from their baseplates. The TPBARs are deposited into a consolidation canister (up to 300 TPBARs per canister). The loaded canister is transferred back into the spent fuel pool for short-term storage until ultimately being placed into shipping casks for transport off site. The TPBAR consolidation canister loading concept has been successfully demonstrated at Department of Energy's Savannah River Site facility. The completed TCF and tools will be tested prior to delivery and also after installation to verify proper operation prior to actual use.

Consolidation Sequence:

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

Approximately 30 days after refueling is complete, TPBAR consolidation begins. The canisters (see 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 in the spent fuel pool and moved to the consolidation fixture using the TPBAR assembly handling tool suspended from the spent fuel pit (SFP) bridge crane. A TPBAR release tool is then utilized by personnel on the platform to detach individual TPBARs from the baseplate. The TPBAR slides along frame guides, through a funnel and into a roller brake, to limit its velocity, and then into the consolidation canister. The funnel, roller brake assembly, and canister are angled

at approximately 15 degrees to enable the TPBARs to stack efficiently into the canister to maximize the loading. Activities take place underwater at a safe shielding water depth.

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

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

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

4. Fuel Handling and Storage for Assemblies Containing TPBARs

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

5. TPBAR Assembly Handling for Consolidation

   The weight of a TPBAR assembly is comparable to a burnable poison rod assembly (BPRA). The configuration of the baseplate and TPBAR attachment details are compatible with existing fuel assemblies and the BPRA handling tool. Therefore, the TPBAR assembly can be

handled with the existing BPRA tool or any other tooling designed for the BPRA's. A postulated drop of the light weight, base plate with TPBARs, within the spent fuel pool/cask load pit area, is bounded by the analysis of a fuel handling accident damaging an irradiated fuel assembly and 24 included TPBARs.

6. TPBAR Consolidation Canister Handling

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

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

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

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

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

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

         Additionally, analysis has been performed to demonstrate that damage to more than 24 TPBARs contained in a canister is precluded for all credible impact scenarios during canister handling. This analysis does not analyze a fuel assembly falling onto a loaded consolidation

canister located in a spent fuel rack. Accordingly, administrative and/or design features will be in place to preclude the possibility of damage to TPBARs loaded into canisters resulting from a fuel handling accident.

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

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

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

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

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

A. The platform (or platform sections) weigh substantially less than ½ of the hook capacity of 125 or 10 tons (Note: The platform is handled as a

single unit, and in two sections during assembly). Along with other design and administrative features, this crane is considered equivalent single-failure-proof consistent with the requirements of NUREG-0612 and NUREG-0554 for this lift.

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

8. TPBAR Transport Cask Handling

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

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

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

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

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

paths, crane inspection and operator training, etc., remain in force.

9. Worker Radiation Exposure During TPBAR Consolidation Activities

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

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

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Sequoyah Nuclear Plant (SQN) Units 1 and 2 in accordance with the proposed changes to the technical specifications (TSs) 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).

This determination evaluates the acceptability in the TS to lower the range of the source range monitors, increase the boron concentration requirements for the cold leg injection accumulators and the refueling water storage tanks (RWSTs), delete requirements for storage of spent fuel in the cask pit pool that is no longer to be utilized, and revise the storage requirements for spent fuel assemblies in the spent fuel pool that have been utilized to produce tritium. Additionally, the TS limit for the total number of tritium producing burnable absorber rods (TPBARs) that can be placed in the core is evaluated. The final change involves the addition of a TPBAR consolidation activity.

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

1. TS Table 3.3-9 - Remote Shutdown Monitoring Instrumentation - Revised Source Range Monitor Range

   The backup source range monitors are for indication of unit shutdown conditions only and do not perform any trip or mitigation functions. The monitors are

not active components such that they could initiate a postulated accident and are not considered a contributor to accident generation. Therefore, the lowering of the indication range for this monitor will not increase the probability of an accident.

Since the monitor has only an indication function, it does not serve to mitigate postulated accidents. While the indications from this monitor can help to identify changing core conditions and promote actions to prevent undesired conditions, this is not a mitigation function credited in the accident analysis and is considered a diverse capability of the plant instrumentation system. Therefore, the proposed change will not impact any credited accident mitigation functions, and by improving shutdown monitoring capability, will not increase the consequences of an accident.

2. TS 3/4.5.1 - Cold Leg Injection Accumulators - 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 added post-LOCA long-term cooling problems associated with boron precipitation in the core following a large break LOCA (LBLOCA).

An evaluation of the non-LOCA events shows that the accumulators do not actuate. An increase in accumulator boron concentration would have no effect on either the steam line break (SLB) at hot zero power event, the feedwater line break event, or the spurious operation of safety injection (SI) system event (events in which an SI signal does occur). Therefore, there is no increase in consequences of the non-LOCA events associated with the proposed increase in accumulator boron concentration.

The accumulators are not assumed to actuate in the steam generator tube rupture (SGTR) event analysis, and the SLB mass and energy (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 in accumulator boron concentration

will not increase the radiological consequences of an accident previously evaluated in the UFSAR.

3. TS 3/4.5.5 - Refueling Water Storage Tank - 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 RWST boron concentration 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. 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 EOPs. Thus, there will be no added post-LOCA long-term cooling problems associated with boron precipitation in the core following a LOCA.

An evaluation of the non-LOCA events indicates that an SI initiation occurs in the SLB at hot zero power event, the feedwater line break event, and

the spurious operation of the SI system event. An increase in the RWST boron concentration would effectively reduce the return to power subsequent to a SLB. Boration is not credited in the feedwater line break analysis and the proposed boron increase is conservatively bounded by the boron inputs to the spurious SI system operation analysis. Therefore, there is no increase in consequences of the non-LOCA events associated with the proposed increase in RWST boron concentration.

The SLB M&E release evaluation relies on control rods for shutdown margin and assumes a minimum boron concentration. For the SGTR, the boron concentration in the accumulators and the RWST are not modeled. 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 in RWST boron concentration will not increase the radiological consequences of an accident previously evaluated in the UFSAR.

4. TS 3/4.7.14 and Bases - Cask Pit Pool Minimum Boron Concentration - Deletion of Requirements

   This change removes the provisions that allow and support the storage of spent fuel in the cask pit pool. By eliminating this provision, the potential for criticality events associated with stored fuel in the cask pit pool is no longer credible. Not

having boron concentration requirements for the cask pit for storage considerations is acceptable based on the removal of TS provisions that would allow such storage. The boron concentration requirement is not considered a contributor to accident generation and therefore, this deletion does not increase the potential for accident generation because spent fuel will not be stored in this location. Likewise, the consequences of an accident will not be increased because the dose generation source, in the form of spent fuel stored in the cask pit, will not be allowed.

5. TS 5.3.1 - Design Features/Reactor Core/Fuel Assemblies

   The insertion of TPBARs into the SQN 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 SQN Units 1 and 2 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 tritium production core (TPC) topical, the TPBAR design is robust to all accident conditions except the large break LOCA (LBLOCA) 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 LBLOCA. This increase was found to be approximately 1495 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 with the

exception of a LBLOCA and a 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.

6. TS 5.6 and TS 3/4.7.13 Bases - Design Features/Fuel Storage and Spent Fuel Pool Minimum Boron Concentration - Revised Storage Requirements for Fuel Assemblies Containing TPBARs

   A specified amount of soluble boron is needed in the spent fuel pool to provide margin to criticality sufficient to mitigate the effects of the most serious spent fuel pool accident condition. Previous spent fuel pool criticality safety analyses (for Type A fuel) determined the required amount of soluble boron to be 700 parts per million (ppm). The new spent fuel pool criticality safety analysis accounting for storage of Type T fuel confirmed that 700 ppm soluble boron still provides the required margin to criticality. Therefore, there is no significant increase in the consequences of previously evaluated accidents postulated for the spent fuel pool. Additionally, the administrative controls for loading the spent fuel pool are not changed and will continue to maintain acceptable storage configurations consistent with the analysis. Therefore, the proposed change will not increase the probability of an accident.

7. 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. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

1. TS Table 3.3-9 - Remote Shutdown Monitoring Instrumentation - Revised Source Range Monitor Range

   The backup source range monitors are for indication of unit shutdown conditions only and do not perform any trip or mitigation functions. The monitors are not active components such that they could initiate a postulated accident and are not considered a contributor to accident generation. Therefore, the lowering of the indication range for this monitor will not create the possibility of a new or different kind of accident.

2. TS 3/4.5.1 - Cold Leg Injection Accumulators - 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 increase in accumulator boron concentration does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. TS 3/4.5.5 - Refueling Water Storage Tank - 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 increase in RWST boron concentration does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4. TS 3/4.7.14 and Bases - Cask Pit Pool Minimum Boron Concentration - Deletion of Requirements

   This change removes the provisions that allow and support the storage of spent fuel in the cask pit pool. By eliminating this provision, the potential for criticality events associated with stored fuel in the cask pit pool is no longer credible. The boron concentration requirement for the cask pit pool is not considered a contributor to accident generation and therefore, this deletion does not increase the potential for accident generation because spent fuel will not be stored in this location.

5. TS 5.3.1 - Design Features/Reactor Core/Fuel Assemblies

   TPBARS have been designed to be compatible with existing fuel assemblies supplied by Framatome-ANP and its predecessor Framatome Cogema Fuels and with conventional Burnable Poison Rod Assembly (BPRA) handling tools, equipment, and procedures. 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 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 with the exception of a LBLOCA and 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, 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.

6. TS 5.6 and TS 3/4.7.13 Bases - Design Features/Fuel Storage and Spent Fuel Pool Minimum Boron Concentration - Revised Storage Requirements for Fuel Assemblies Containing TPBARs

   The storage in the spent fuel pool of spent fuel that has contained TPBARs is not a fundamental change in the use of the spent fuel pool. Specific provisions have been made for burnup and cooling time requirements in allowable configurations to ensure safe storage. The same administrative program to control storage requirements in the spent fuel pool will be utilized to handle Type A and Type T spent fuel. Therefore, the possibility of a new or different accident than previously evaluated has not been created.

7. TPBAR Consolidation Activity

   The consolidation and handling systems are designed to preclude the possibility of a consolidating and/or handling event which could damage more than 24 TPBARs. 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 Table 3.3-9 - Remote Shutdown Monitoring Instrumentation - Revised Source Range Monitor Range

   The backup source range monitors are for indication of unit shutdown conditions only and do not perform

any trip or mitigation functions. The lowering of the monitor's range does allow improved indication of core conditions with the TPCs. While this monitor does not have any trip or accident mitigation functions, this change will improve the ability to assess the conditions of the unit such that necessary actions can be initiated to prevent undesired conditions. Therefore, the proposed change will not reduce a margin of safety.

2. TS 3/4.5.1 - Cold Leg Injection Accumulators - 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 small break LOCA (SBLOCA) analysis 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 LBLOCA PCT; therefore, the ECCS acceptance criteria limit, dictated by 10 CFR 50.46, 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 long-term cooling problems associated with boron precipitation in the reactor vessel and core. The licensing analyses for containment, equipment qualification, and environmental consequences remain bounding and applicable and the acceptance criteria of the related events continue to be met. The proposed increase in accumulator boron concentration, therefore, does not involve a significant reduction in a margin of safety.

3. TS 3/4.5.5 - Refueling Water Storage Tank - 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 analysis 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 LBLOCA PCT; therefore, the ECCS acceptance criteria limit, dictated by 10 CFR 50.46, 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 analyses for containment, equipment qualification, and environmental consequences remain bounding and applicable and the acceptance criteria of the related events continue to be met. The proposed increase in RWST boron concentration, therefore, does not involve a significant reduction in a margin of safety.

4. TS 3/4.7.14 and Bases - Cask Pit Pool Minimum Boron Concentration - Deletion of Requirements

   This change removes the provisions that allow and support the storage of spent fuel in the cask pit pool. This change will not alter plant systems, operating methods, or plant setpoints that maintain the margin of safety. Boron concentration will continue to be properly maintained for the storage of spent fuel in the spent fuel pool as required by the analysis to control inadvertent criticality events. Therefore, this change will not reduce the margin of safety.

5. TS 5.3.1 - 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.

6. TS 5.6 and TS 3/4.7.13 Bases - Design Features/Fuel Storage and Spent Fuel Pool Minimum Boron Concentration - Revised Storage Requirements for Fuel Assemblies Containing TPBARs

   Addition of fuel assemblies containing TPBARs to the spent fuel pool is consistent with the pool design function. Specific provisions have been made as a result of reanalysis of spent fuel pool criticality safety analysis to limit storage configurations and burnup or cooling time requirements to those that will provide for safe storage of fresh and spent fuel. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

7. TPBAR Consolidation Activity

The changes do not affect the safety-related performance of any plant operations, system, structures, or components. Therefore, there is no significant reduction in the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The environmental impacts of producing tritium in TVA's Sequoyah Units 1 and 2 were assessed in a 1999, "Final Environmental Impact Statement (EIS) for the Production of Tritium in a Commercial Light Water Reactor," (DOE/EIS-0288) prepared by the Department of Energy. 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
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NO. 327, 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
MARKED PAGES

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II. MARKED PAGES

See attached

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

INDEX

DESIGN FEATURES

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

	INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Source Range Nuclear Flux	NOTE 1	~~1 to 1 x 10^6~~ cps ***0.1 to 1 x 10^5***	1
2.	Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3.	Reactor Coolant Temperature - Hot Leg	NOTE 1	0-650°F	1/loop
4.	Pressurizer Pressure	NOTE 1	0-3000 psig	1
5.	Pressurizer Leve1	NOTE 1	0-100%	1
6.	Steam Generator Pressure	NOTE 1	0-1200 psig	1/steam generator
7.	Steam Generator Leve1	NOTE 2 or near Auxilary F. W. Pump	0-100%	1/steam generator
8.	Deleted			
9.	RHR Flow Rate	NOTE 1	0-4500 gpm	1
10.	RHR Temperature	NOTE 1	50-400°F	1
11.	Auxiliary Feedwater Flow Rate	NOTE 1	0-440 gpm	1/steam generator

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained ~~borated~~ water volume of between 7615 and 7960 gallons of borated water,

c. Between ~~2400~~ 3500 and ~~2700~~ 3800 ppm of boron,

d. A nitrogen cover-pressure of between 624 and 668 psig, and

e. Power removed from isolation valve when RCS pressure is above 2000 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

----

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume ~~of between 370,000~~ and 375,000 gallons,

b. A boron concentration of between ~~2500~~ 3600 and ~~2700~~ 3800 ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

[This page deleted]

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The cask pit pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: Whenever fuel assemblies are stored in the cask pit rack.

ACTION:

a. With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and initiate action to restore cask pit pool boron concentration to within limit. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.1 Verify at least once per 7 days the cask pit pool boron concentration is within limit.

4.7.14.2 Verify at least once per 72 hours during fuel movement the cask pit pool boron concentration is within limit and until the configuration of the assemblies in the storage rack is verified to comply with the criteria loading criteria specified in Design Feature 5.6.1.1.d.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-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. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the Unit 1 Operating Cycle 12.

Insert

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

Sequoyah is authorized to place a maximum of 2256 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

[Insert 1]

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

a. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron.*

b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Arrangements of one or more of three different arrays (Regions) or sub-arrays as illustrated in Figures 5.6-1 and 5.6-1a. These arrangements in the spent fuel storage pool have the following definitions:

Type A

1. Region 1 is designed to accommodate new fuel ~~with a~~ maximum enrichment of 4.95 ± 0.05 wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh ~~assembly~~ with 3 spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 5.6-2 and defined by the ~~equations in Table 5.6-1. Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.~~ The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-2 or Table 5.6-1.

   Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or integral fuel burnable absorber (IFBA). For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 5.6-2.

   Restrictions in Region 1

   Any of the three sub-arrays illustrated in Figure 5.6-1a may be used in any combination provided that:

   ~~1~~ ***A)*** Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-1 and 5.6-2, as appropriate.

   ~~2~~ ***B)*** The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

   ~~3~~ ***C)*** If Region 1 arrays are used in conjunction with ~~Region 2~~ ***3*** ⇑ or Region ~~3~~ ***4*** ⇑ arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 5.6-1).

---

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

Insert 1

For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel that has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

5.6 FUEL STORAGE

[Insert 2]

2. Region 2 is designed to accommodate ***[Insert 3]*** fuel of 4.95±0.05 wt% U-235 initial enrichment burned to at least 30.27 ***[Insert 4]*** 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 in MWD/KgU and cooling time is given by the equations in Table 5.6-3 ***[Insert 5]*** ~~in terms of E, where E is the initial enrichment in the axial zone of highest enrichment (wt% U-235)~~. The minimum required burnups are illustrated in Figure 5.6-3 ***[Insert 5]*** in terms of the initial enrichment and cooling time.

Restrictions in Region 2

The following restrictions apply to the storage of spent fuel in the Region 2 cells:

~~1~~ ***A***) The spent fuel shall conform to the minimum burnup requirements defined by the equations in Table 5.6-3 ***[Insert 6]***. Linear interpolation between cooling times may be made if desired.

~~2~~ ***B***) For the interface with Region 1 ***[Insert 7]***storage cells, fresh fuel in Region 1 ***[Insert 7]***shall not be stored adjacent to spent fuel assemblies in the Region 2 storage cells.

[Insert 8]

3. Region 3 is designed to accommodate fuel of 4.95±0.05 wt% U-235 initial enrichment (or fuel assemblies of any lower reactivity) in a 2-out-of-4 checkerboard arrangement with water-filled cells. The water-filled cells shall not contain any components bearing any fissile material, but may accommodate miscellaneous items or equipment.

Restrictions in Region 3

[Insert 9] ***3*** ***1 or 4***

~~1~~ ***A***) For the interface between Region ~~1~~ ⇑ and Region ~~3~~ ⇑ storage regions, fresh fuel assemblies shall not be stored adjacent to each other.

non-fissile bearing

~~2~~ ***B***) If miscellaneous ⇑ items or equipment are stored in the water cells of Region 3, the total volume of the miscellaneous items shall be no more than 75% of the storage cell volume.

loose

~~3~~ ***C***) No ⇑ fuel rods~~, assemblies,~~ or items containing fissile material shall be stored in the water cells of Region 3.

[Insert 10]

~~An empty cell is less reactive than any cell containing fuel and therefore may be used as a Region 1, Region 2, or Region 3 cell in any arrangement.~~

~~d. Region 2 array described above may be used in the 15 x 15 storage rack module in the cask loading area of the cask pit.~~

[Insert 11]

e. A nominal concentration of 2000 ppm boron ~~*is*~~ in the pool water. This concentration of soluble boron provides a margin sufficient to allow timely detection of a boron dilution accident and corrective action before the minimum concentration (700 ppm) required to protect against the most severe postulated fuel handling accident or before the minimum concentration (300 ppm) required to maintain the storage configuration design basis (k_{eff} less than 0.95) is reached.

Insert 2

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 1, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

Insert 3

Type A or Type T

Insert 4

(Type A) or 33.1095 (Type T)

Insert 5

(Type A) or 5.6-4 (Type T)

Insert 6

or 5.6-4, as appropriate

Insert 7

or 4

Insert 8

C) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 2, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

Insert 9

The following restrictions apply to the storage of fuel in the Region 3 cells:

Insert 10

4. Region 4 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 $\pm$ 0.05 wt% U-235 (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with three Type T spent fuel assemblies having burnup and cooling times illustrated in Figure 5.6-5 and defined by the equations in Table 5.6-5. The presence of either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-5 or Table 5.6-5.

Insert 10 continued

One alternative storage array (or sub-array) is acceptable in Region 4 if the fresh fuel contains rods with gadolinia fuel burnable absorber. For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array is defined by the equations in Table 5.6-6 and illustrated in Figure 5.6-6. For fresh assemblies containing more than eight (8) gadolinia bearing fuel rods, the limiting burnup for eight (8) gadolinia rods shall apply.

Restrictions in Region 4

Any of the two sub-arrays illustrated in Figure 5.6-1a applying to Region 4 storage may be used in any combination provided that:

A) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-5 and 5.6-6, as appropriate.

B) The arrangement of Region 4 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

C) If Region 4 arrays are used in conjunction with Region 1 or 3 arrangements, the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see Figure 5.6-1)

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 4, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

Insert 11

d. An empty cell (or a cell containing non-fissile bearing miscellaneous items displacing no more than 75% of the storage cell volume) is less reactive than any cell containing fuel and therefore may be used as a Region 1, 2, 3, or 4 cell in any arrangement.

5.6 FUEL STORAGE

5.6-7

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed for fuel enriched to 5.0 ~~weight~~ percent U-235 and shall be maintained with the arrangement of 146 storage locations shown in Figure ~~5.6-4~~. The cells shown as empty cells in Figure ~~5.6-4~~ shall have physical barriers installed to ~~ensure~~ that inadvertant loading of fuel assemblies into these locations does not occur. This configuration ensures k_{eff} will remain less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 under optimum moderation conditions.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

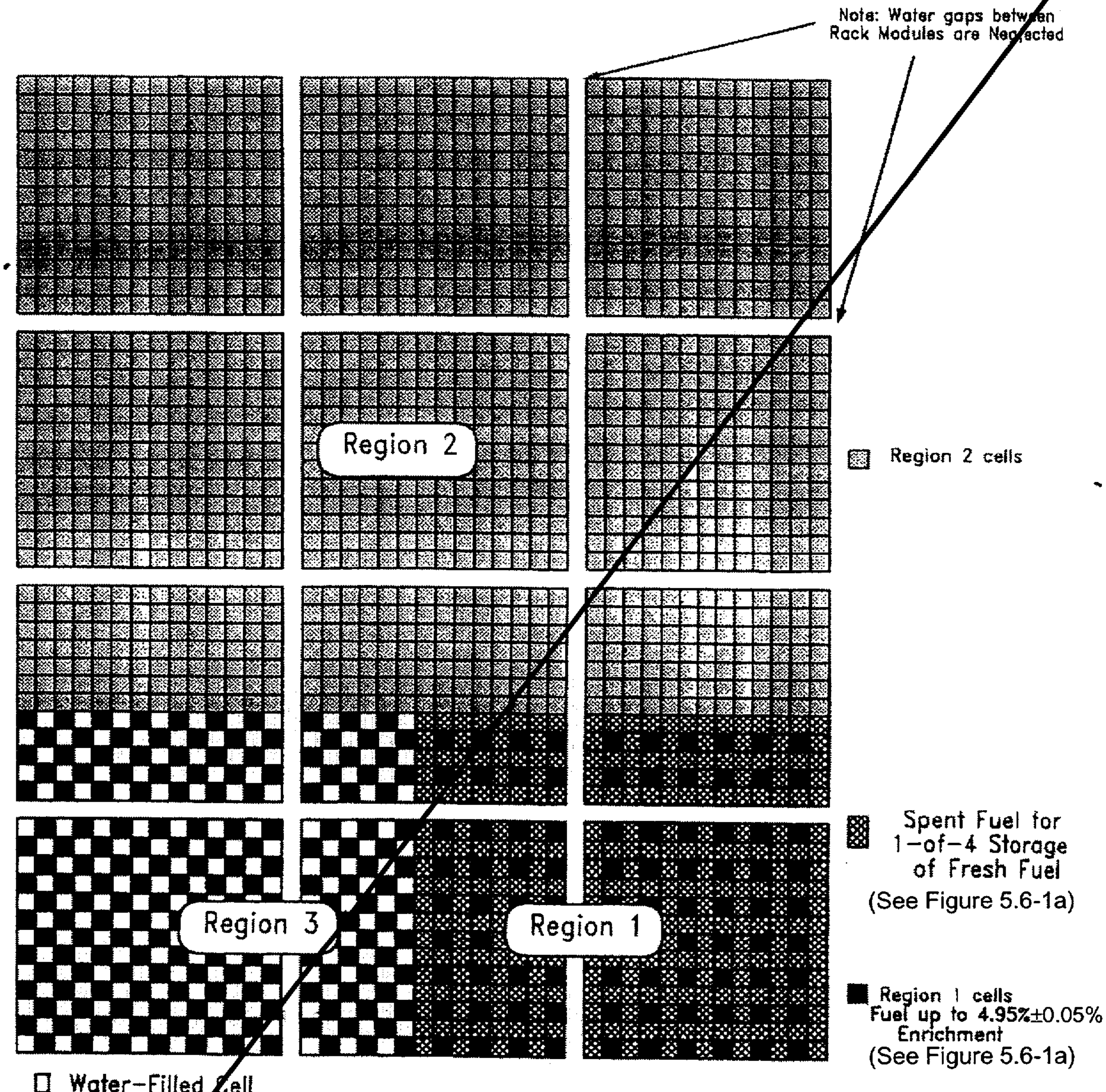
CAPACITY

5.6.3 The spent fuel storage pool is ~~designed and shall be maintained with~~ a ~~storage~~ capacity limited ~~to~~ no more than 2091 fuel assemblies. ~~In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.~~

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

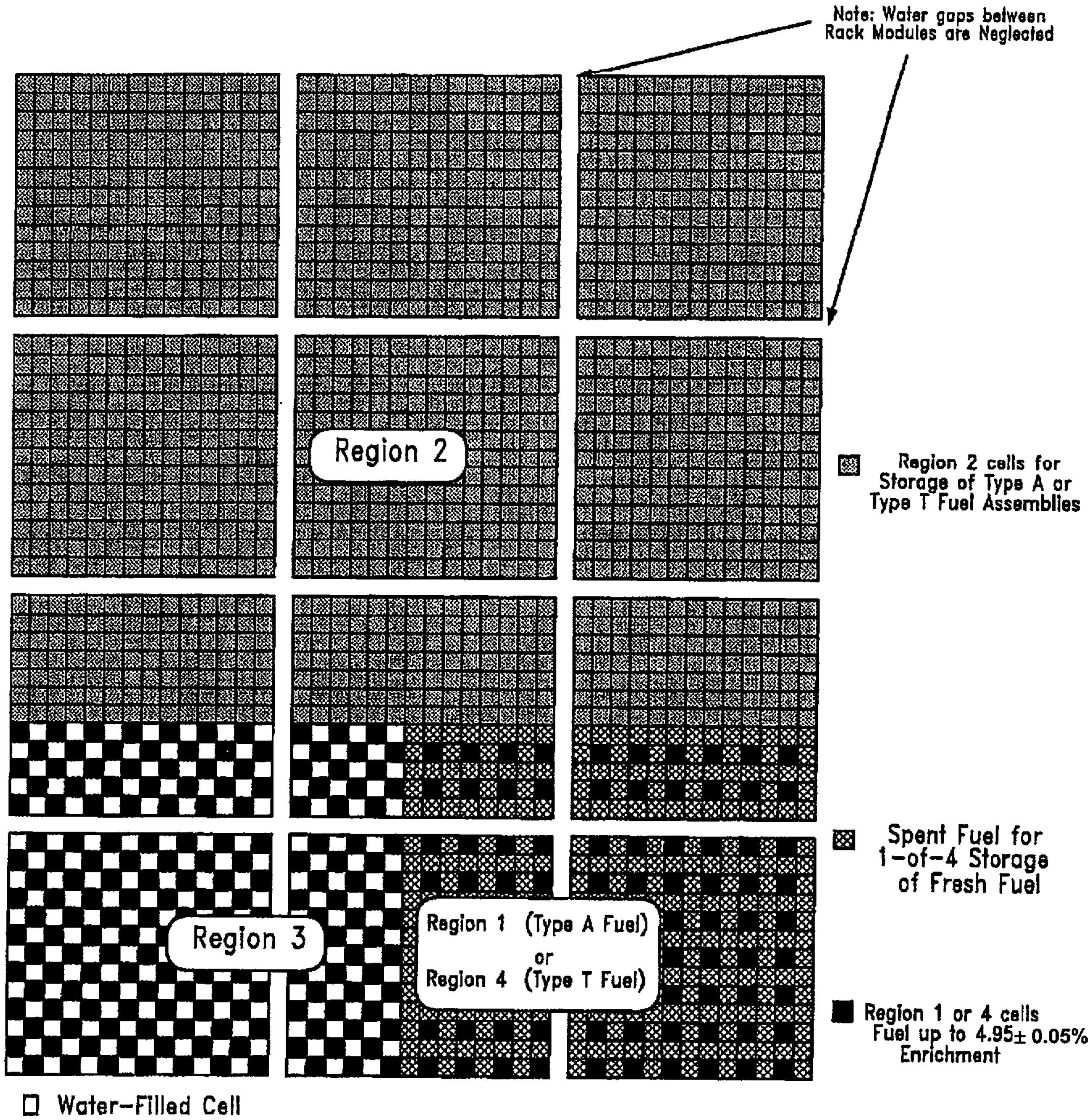
[Replace with Insert 12]



Note: The edges of the sketch above are not necessarily the edges of the pool. The Regions may appear anywhere in the pool and in any orientation, subject to the restriction in Design Feature 5.6.1.1.c.

Figure 5.6-1
Arrangements of Fuel Storage Regions in the Sequoyah Spent Fuel Storage Pool

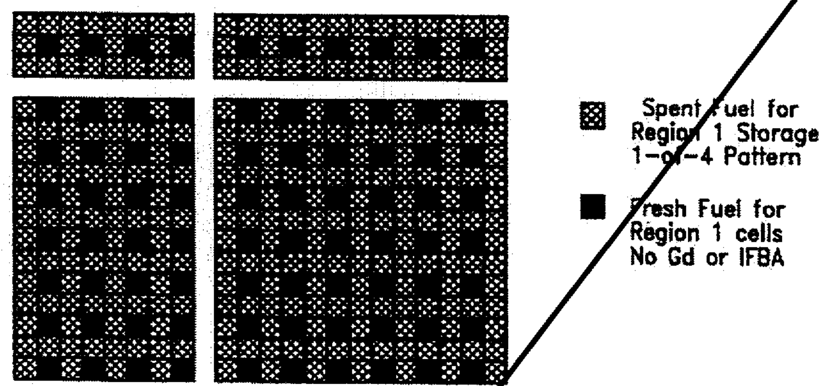
Insert 12



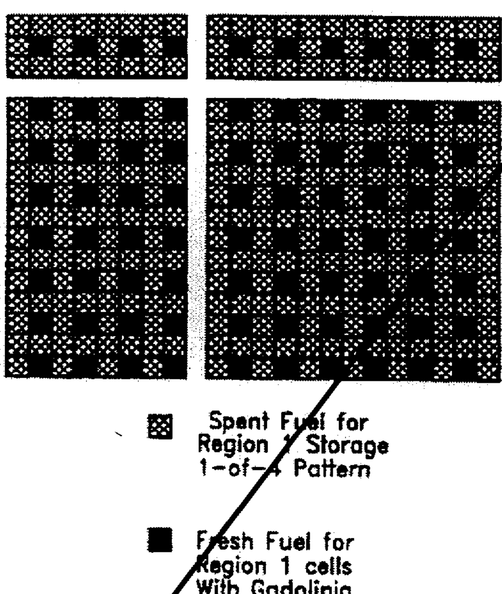
Note: The edges of the sketch above are not necessarily the edges of the pool. The Regions may appear anywhere in the pool and in any orientation, subject to the restrictions in Design Features 5.6.1.1.c.

FIG 5.6-1 Arrangements of Fuel Storage Regions in the Sequoyah Spent Fuel Storage Pool

[Replace with Insert 13]



NOTE: WHEN CREDIT IS TAKEN FOR GADOLINIA OR IFBA RODS IN FRESH ASSEMBLIES THE SPENT FUEL ASSEMBLIES NEED NOT HAVE CONTAINED GADOLINIA OR IFBA RODS..



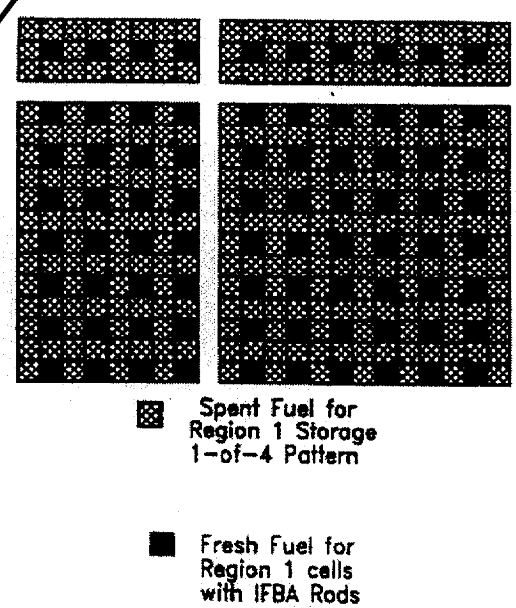
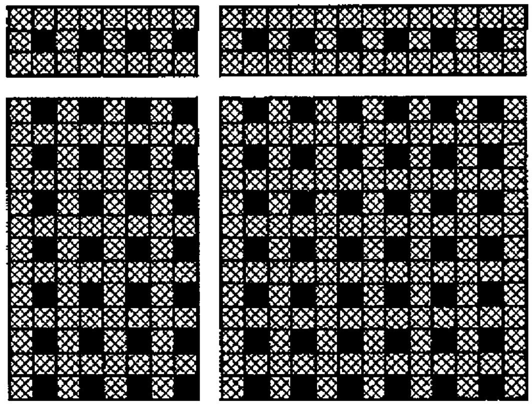


Figure 5.6-1a
Acceptable Spent Fuel Pool Loading Patterns for Checkerboard Storage of Fresh and Spent Fuel Assemblies - Example

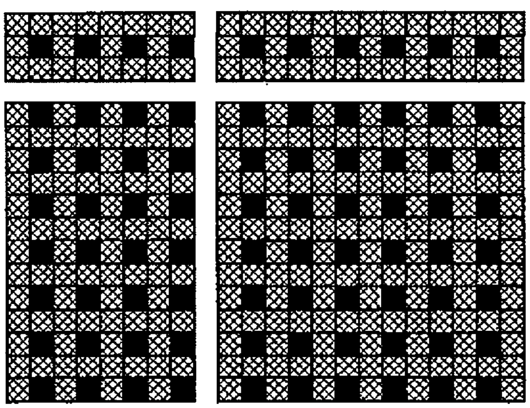
Insert 13



Spent Fuel for Storage in Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

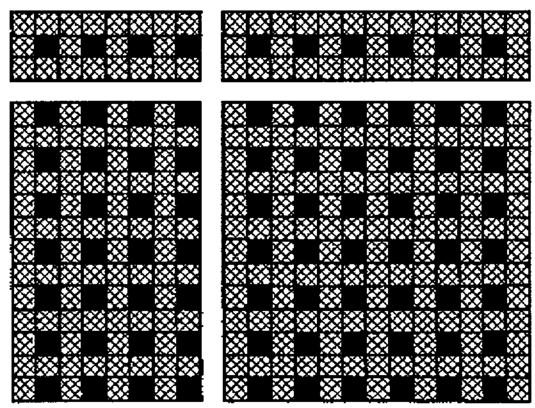
Fresh Fuel for Region 1 or Region 4 cells (No Gd or IFBA)

NOTE: WHEN CREDIT IS TAKEN FOR GADOLINIA RODS IN FRESH ASSEMBLIES THE SPENT FUEL ASSEMBLIES NEED NOT HAVE CONTAINED GADOLINIA RODS..



Spent Fuel for Storage of Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

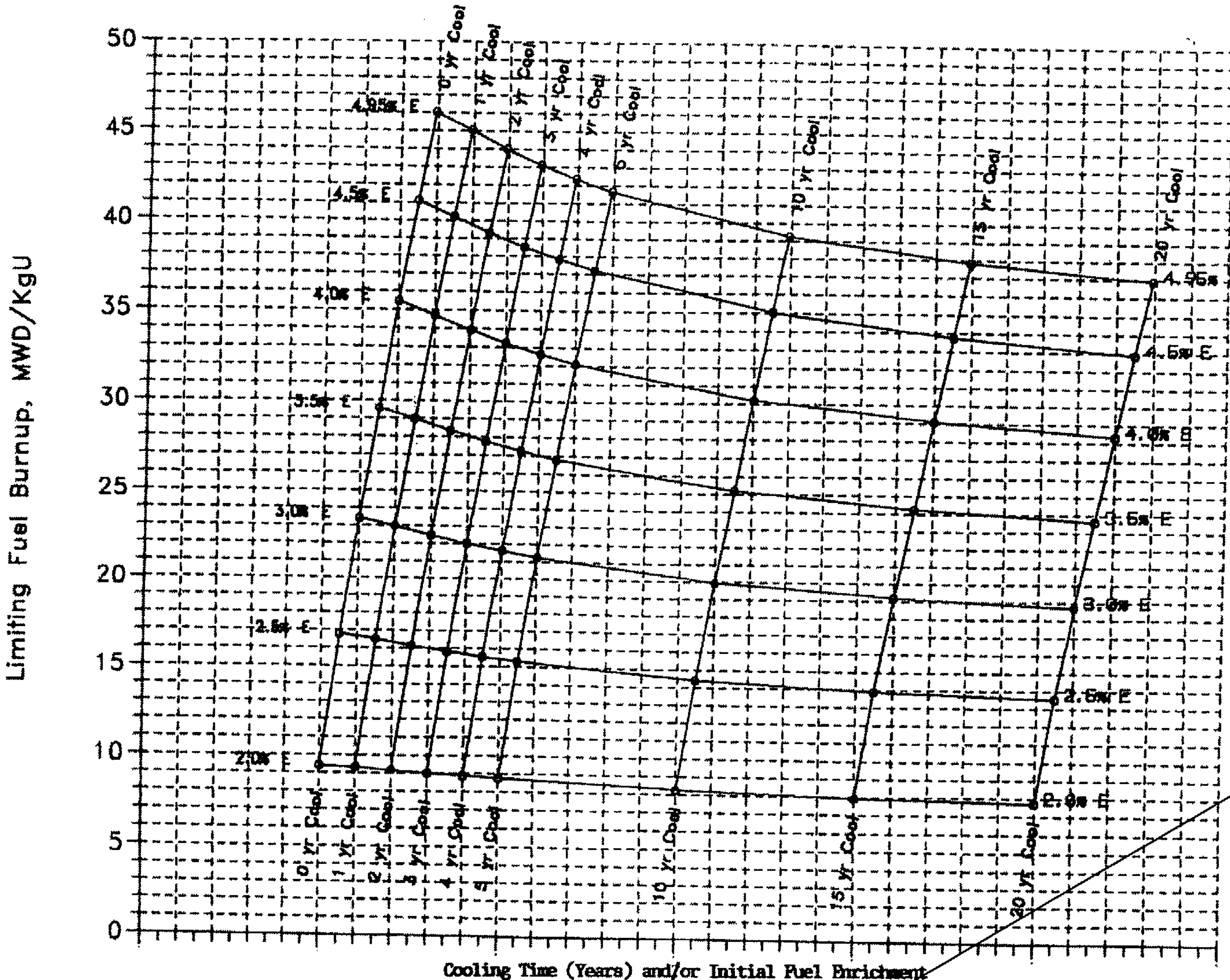
Fresh Fuel with Gadolinia for Region 1 or Region 4 cells



Spent Fuel for Storage in Region 1 in 1-of-4 Pattern

Fresh Fuel with IFBA Rods for Storage in Region 1 only (1-of-4 Pattern)

Fig. 5.6-1a Acceptable Storage Patterns for Checkerboard Storage of Fresh and Spent Fuel Assemblies in Region 1 or Region 4 – Example



for Type A Fuel

Fig. 5.6-2 3-Dimensional Plot of Minimum Fuel Burnups in Region 1 for Enrichments and/or Cooling Times

For Type A Fuel

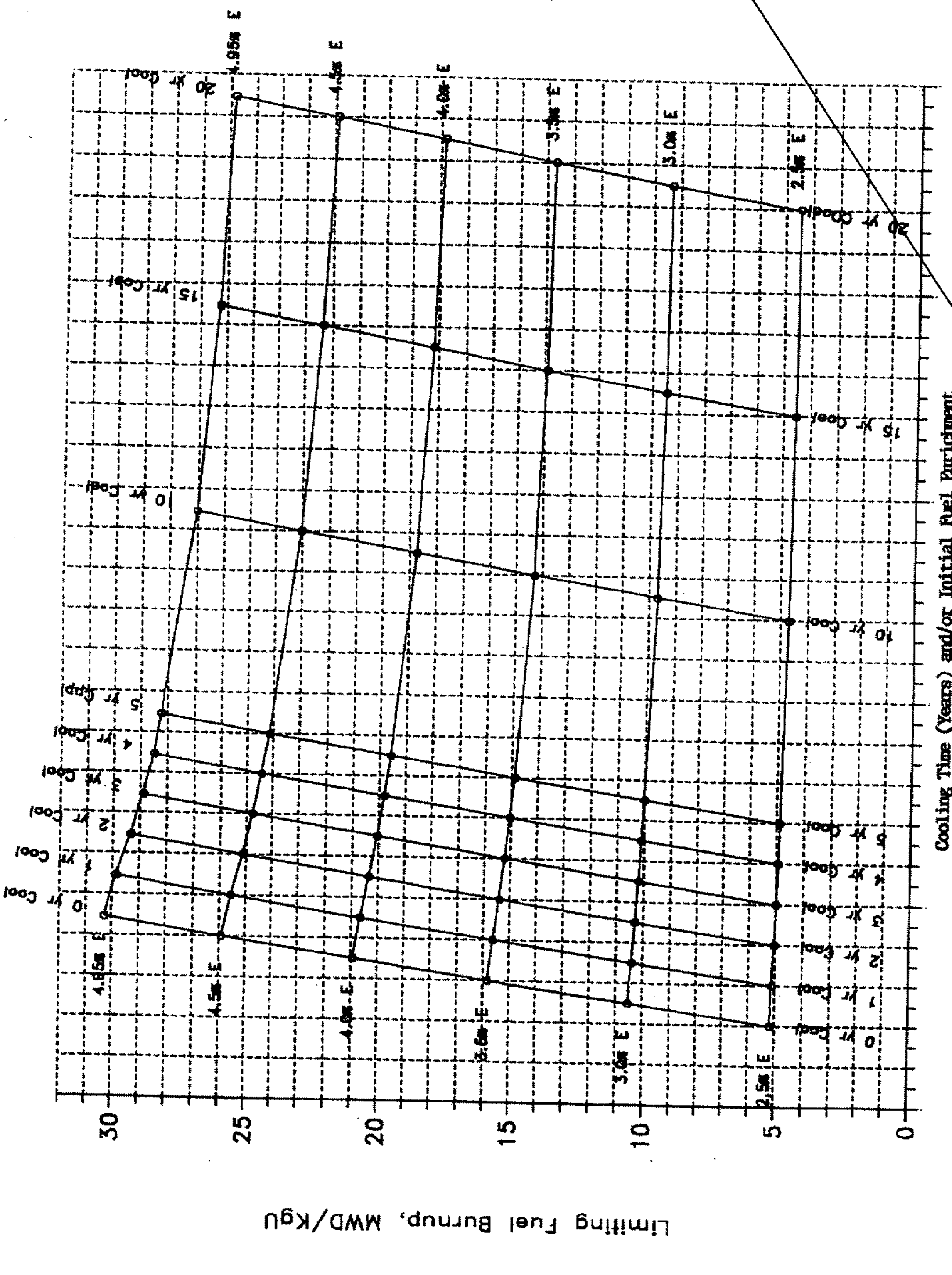
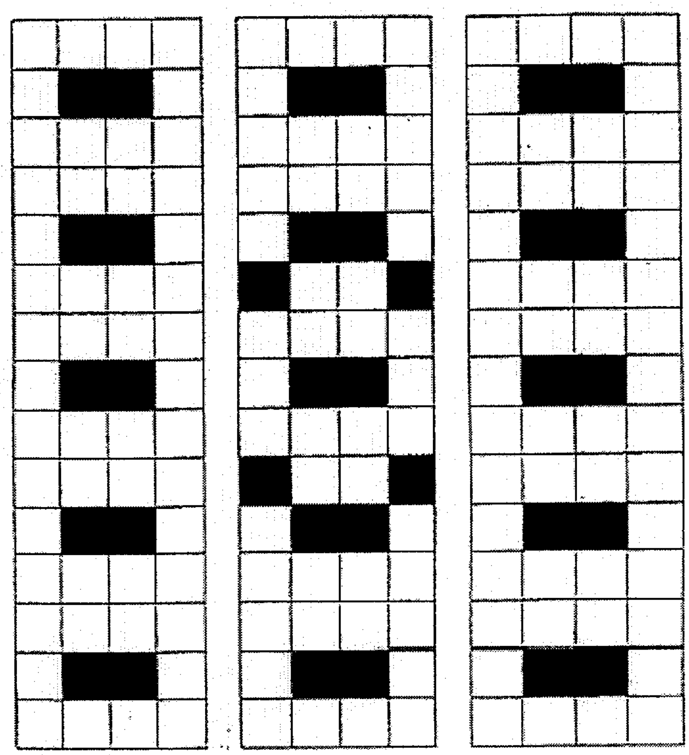


Fig. 5.6-3 3- Dimensional Plot of Minimum Fuel Burnups in Region 2 for Enrichments and Cooling Times

[Add Inserts 14, 15, and 16]



Basic Cell 21 inch X 21 inch

Empty Cell

9 - 4 X 5 Cell Racks

146 / 180 Loading Pattern

Figure ~~5.6-4~~ ***5.6-7***
New Fuel Pit Storage Rack Loading Pattern

Insert 14

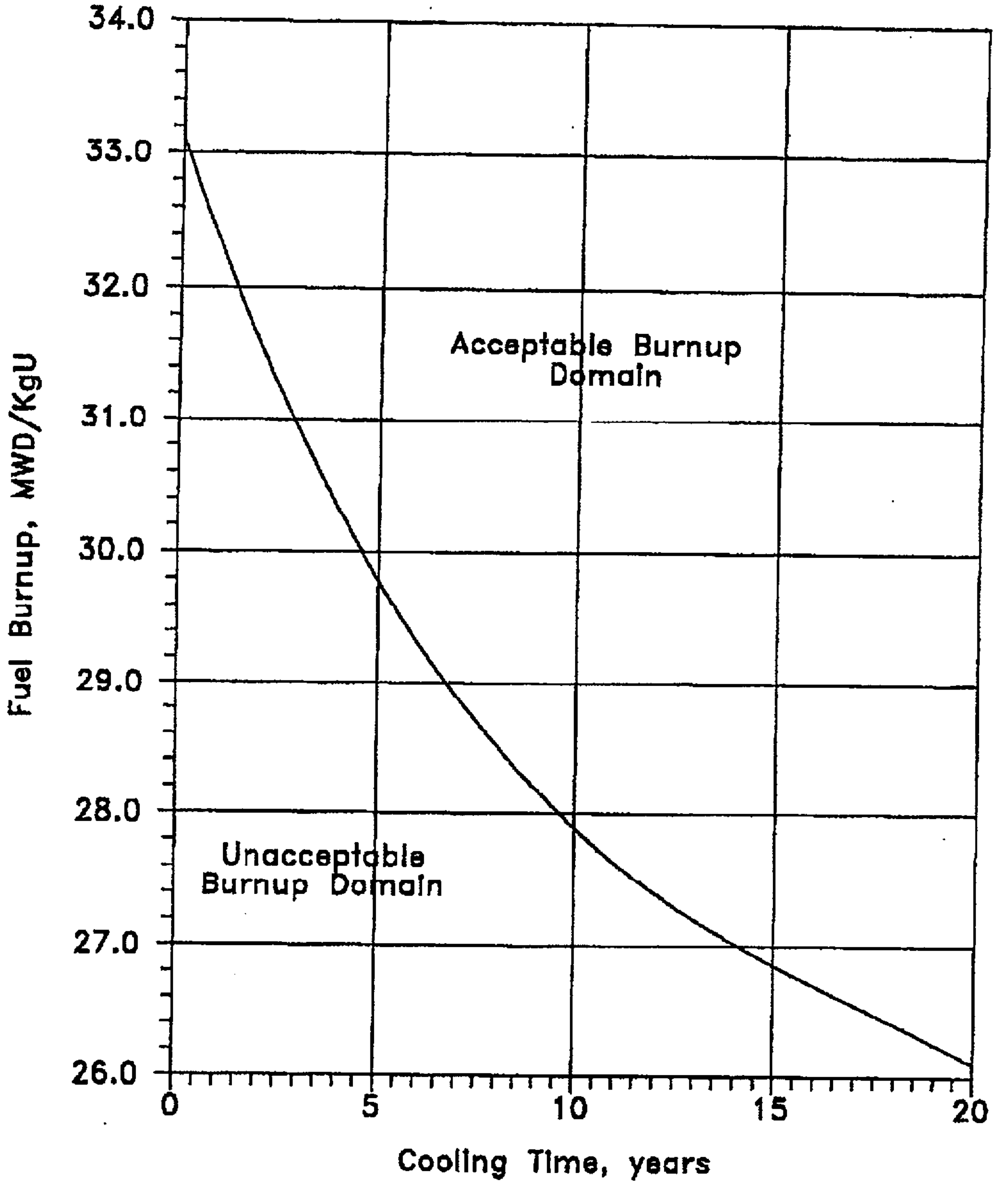


Fig. 5.6–4 Limiting Burnup Requirements in Region 2 for Face Adjacent Storage of Type T Spent Fuel Assemblies

Insert 15

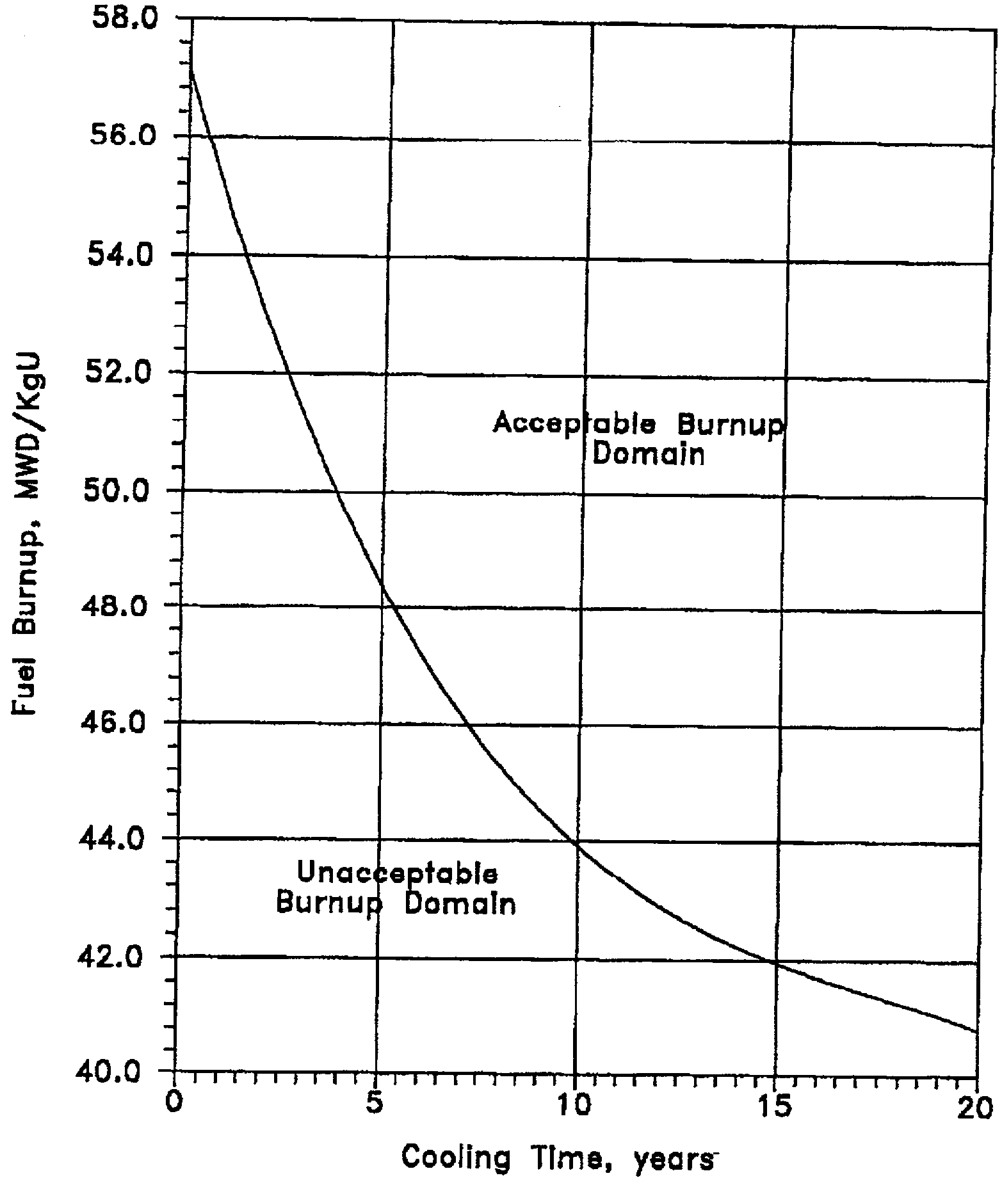


Fig. 5.6-5 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh and 3 Type T Spent Fuel Assemblies

Insert 16

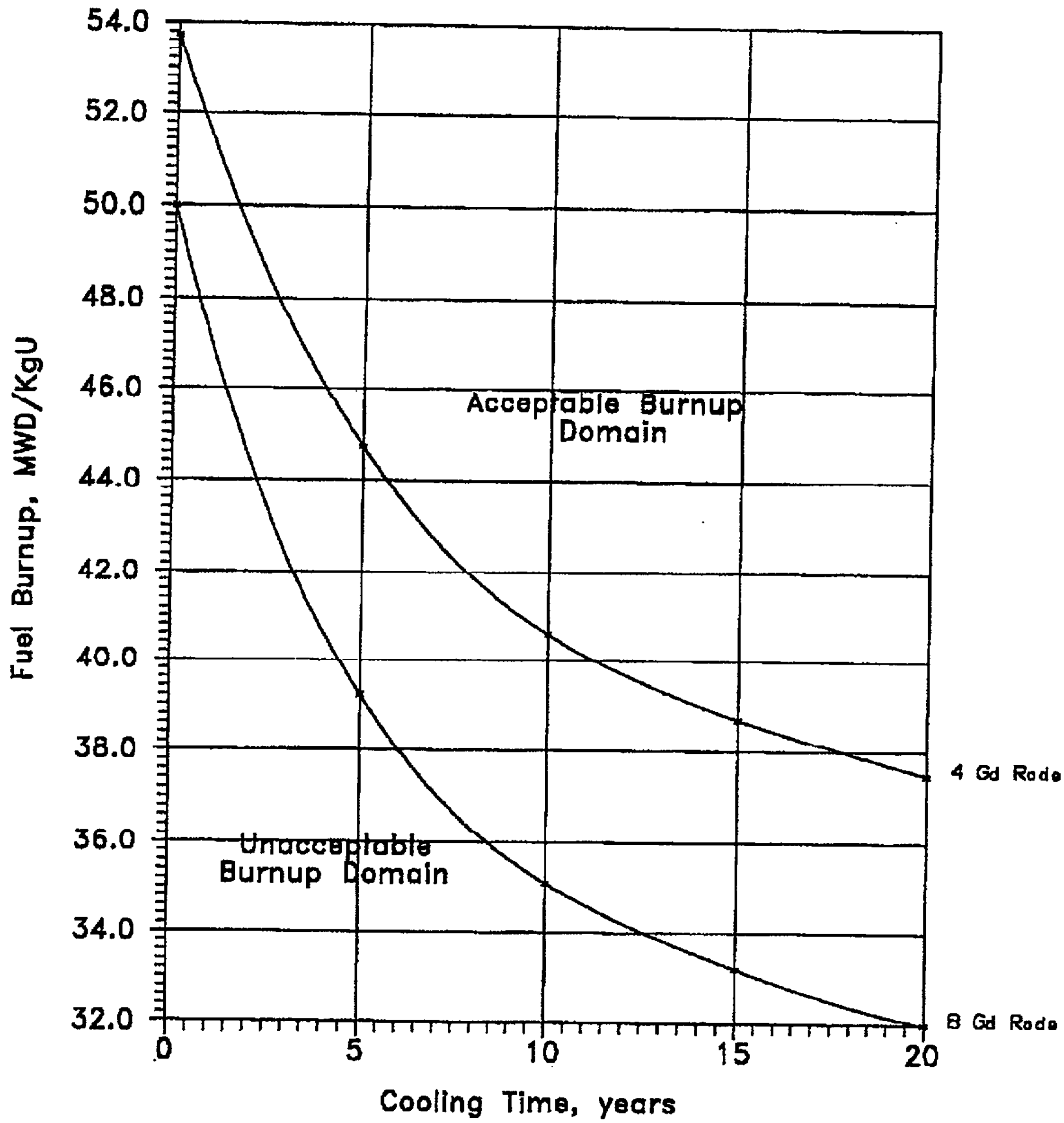


Fig. 5.6-6 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh (with Gadolinia) and 3 Type T Spent Fuel Assemblies

Table 5.6-1
Region 1 Storage Burnup Restrictions: Checkerboard of 1
Fresh Fuel Assembly $\Downarrow$ and 3 ***Type A*** Spent Fuel Assemblies ~~(Without Gadolinium or IFBA Rods)~~
(Without Gadolinium or IFBA Rods)

For Zero Year Cooling Time $Bu\ (limit) = -28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
For One Year Cooling Time $Bu\ (limit) = -27.3317 + 22.5087 \times E - 2.40586 \times E^2 + 0.164207 \times E^3$
For Two Years Cooling Time $Bu\ (limit) = -26.4693 + 21.8404 \times E - 2.31873 \times E^2 + 0.158218 \times E^3$
For Three Years Cooling Time $Bu\ (limit) = -25.7404 + 21.2659 \times E - 2.24287 \times E^2 + 0.153018 \times E^3$
For Four Years Cooling Time $Bu\ (limit) = -25.1367 + 20.7910 \times E - 2.18484 \times E^2 + 0.1499363 \times E^3$
For Five Years Cooling Time $Bu\ (limit) = -24.5981 + 20.3568 \times E - 2.12719 \times E^2 + 0.145431 \times E^3$
For Ten Years Cooling Time $Bu\ (limit) = -23.2050 + 19.2969 \times E - 2.06993 \times E^2 + 0.145875 \times E^3$
For Fifteen Years Cooling Time $Bu\ (limit) = -22.6098 + 18.8544 \times E - 2.08617 \times E^2 + 0.150473 \times E^3$
For Twenty Years Cooling Time $Bu\ (limit) = -22.3017 + 18.622 \times E - 2.11206 \times E^2 + 0.15467 \times E^3$

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-2

Region 1 Storage Burnup Restrictions with Gadolinium or IFBA ***in Fresh Fuel***

Type A

With Gadolinium Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 ⇑ Spent Fuel Assemblies

Zero Year Cooling Time, 0 Gadolinia Rods
$Bu\ (limit) = -28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 4 Gadolinia Rods
$Bu\ (limit) = -28.4012 + 22.0062 \times E - 2.19268 \times E^2 + 0.143601 \times E^3$
Zero Year Cooling Time, 8 Gadolinia Rods
$Bu\ (limit) = -31.4262 + 22.0768 \times E - 2.38845 \times E^2 + 0.164888 \times E^3$

Note: If more that 8 Gadolinium rods per assembly, use the 8 rod correlation

Type A

With IFBA Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 ⇑ Spent Fuel Assemblies

Zero Year Cooling Time, 0 IFBA Rods
$Bu\ (limit) = -28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 16 IFBA Rods
$Bu\ (limit) = -28.5048 + 21.6411 \times E - 2.15262 \times E^2 + 0.140904 \times E^3$
Zero Year Cooling Time, 32 IFBA Rods
$Bu\ (limit) = -31.0949 + 22.0435 \times E - 2.36088 \times E^2 + 0.162229 \times E^3$
Zero Year Cooling Time, 48 IFBA Rods
$Bu\ (limit) = -33.1342 + 22.3999 \times E - 2.55367 \times E^2 + 0.18082 \times E^3$
Zero Year Cooling Time, 64 IFBA Rods
$Bu\ (limit) = -36.0468 + 24.1492 \times E - 3.11807 \times E^2 + 0.233987 \times E^3$

Note: If more that 64 IFBA rods per assembly, use the correlation for 64 IFBA rods

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-3
Region 2 Storage Burnup Restrictions
For Type A Fuel

Zero Cooling Time Bu (limit) = $- 23.8702 + 12.3026 \times E - 0.275672 \times E^2$
1 Year Cooling Time Bu (limit) = $- 23.6854 + 12.2384 \times E - 0.287498 \times E^2$
2 Years Cooling Time Bu (limit) = $- 23.499 + 12.1873 \times E - 0.305988 \times E^2$
3 Years Cooling Time Bu (limit) = $- 23.3124 + 12.1249 \times E - 0.319566 \times E^2$
4 Years Cooling Time Bu (limit) = $- 23.1589 + 12.0748 \times E - 0.332212 \times E^2$
5 Years Cooling Time Bu (limit) = $- 22.6375 + 11.7906 \times E - 0.307623 \times E^2$
10 Years Cooling Time Bu (limit) = $- 21.7256 + 11.3660 \times E - 0.31029 \times E^2$
15 Years Cooling Time Bu (limit) = $- 21.1160 + 11.0663 \times E - 0.306231 \times E^2$
20 Years Cooling Time Bu (limit) = $- 20.6055 + 10.7906 \times E - 0.29291 \times E^2$

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-4

Face Adjacent Storage of Type T Spent Fuel (Region 2)

$$Bu\ (limit) = 33.1095 - 0.845146 \times CT + 0.0399888 \times CT^2 - 0.000762846 \times CT^3$$

Table 5.6-5

Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel (Region 4: 1 Fresh Assembly and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

$$Bu\ (limit) = 57.118 - 2.13277 \times CT + 0.0772537 \times CT^2 + 0.00127446 \times CT^3 - 9.15855\ E\text{-}5 \times CT^4$$

Table 5.6-6

Gadolinia Credit: Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel (Region 4: 1 Fresh Assembly with Gadolinia and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

4 Gadolinia Rods

$$Bu\ (limit) = 53.73 - 2.5265 \times CT + 0.172283 \times CT^2 - 0.00585995 \times CT^3 + 0.0000766655 \times CT^4$$

8 Gadolinia Rods

$$Bu\ (limit) = 50.00 - 3.26817 \times CT + 0.276117 \times CT^2 - 0.0117934 \times CT^3 + 0.000195334 \times CT^4$$

Note: 1. If more than 8 gadolinia rods per assembly, use the 8 rod correlation
2. BU = Fuel Burnup, MWD/Kg-U; CT = Cooling Time of Spent Fuel Assemblies, Years

, and 4) tritium and hydrogen that exist in the Tritium Producing Burnable Absorber Rods prior to the accident

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the hydrogen mitigation system, consisting of 68 hydrogen ignitions per unit, is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, ~~and~~ 3) corrosion of metals within containment. These hydrogen control systems are designed to mitigate the effects of an accident as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", Revision 2 dated November 1978. The hydrogen monitors of Specification 3.6.4.1 are part of the accident monitoring instrumentation in Specification 3.3.3.7 and are designated as Type A, Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The operability of at least 66 of 68 ignitors in the hydrogen mitigation system will maintain an effective coverage throughout the containment. This system of ignitors will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1071 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,082,024 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the

PLANT SYSTEMS

BASES

3/4.7.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION

Reports HI-992349 (Ref 1) and HI-2012629 (Ref 9)

BACKGROUND

The spent fuel racks have been analyzed in accordance with the Holtec International methodology contained in Holtec ~~Report HI - 992349 (Ref. 1)~~. This methodology ensures that the spent fuel rack multiplication factor, k_{eff} is less than or equal to 0.95, as recommended by the NRC ~~guidance contained in NRC Letter~~ to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 and USNRC Internal Memorandum from L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998 (Refs. 2 & 3). The codes, methods, and techniques contained in the methodology are used to satisfy the k_{eff} criterion. The spent fuel storage racks were analyzed using Westinghouse 17x17 V5H fuel assemblies, with enrichments up to 4.95 ±0.05 w/o U-235 and configurations which take credit for checkerboarding, burnup, soluble boron, integral fuel burnable absorbers (such as IFBA or gadolinia), and cooling time to ensure that k_{eff} is maintained ≤ 0.95, including uncertainties, tolerances, and accident conditions. ***[Insert 17]*** In addition, the SFP k_{eff} is maintained < 1.0, including uncertainties, ~~tolerances~~ on a 95/95 basis without any soluble boron. Calculations were performed to evaluate the reactivity of fuel types used at SQN. The results show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity, thereby bounding all fuel types utilized and stored ~~at SQN~~.

Replace with Insert 18 ⇒

~~In the high density Spent Fuel Storage Rack design (Refs. 1 and 4), the spent fuel storage pool is divided into three separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, or spent fuel regardless of the discharge fuel burnup in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with enrichment, burnup and cooling times in accordance with Design Features 5.6.1.1.c.1. Region 2 is designed to accommodate fuel which have 4.95 ± 0.05 wt% U-235 initial enrichment burned to at least 30.27 MWD/KgU (assembly average), or fuel of other enrichment with a burnup yielding an equivalent reactivity in the fuel racks in accordance with Design Features 5.6.1.1.c.2. Region 3 is designed to accommodate fuel of 4.95 ± 0.05 wt% U-235 initial enrichment or fuel assemblies of any lower reactivity in a 2-out-of-4 checkerboard arrangement with water-filled cells and in accordance with Design Features 5.6.1.1.c.3.~~

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 < 1.0 be evaluated in the absence of soluble boron. Hence, the design of all 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 (Ref. 5) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most

(continued)

Insert 17

The analysis also accounts for the reactivity effects of operating the fuel with discrete burnable poisons (such as burnable poison rod absorbers or tritium producing burnable absorber rods).

Insert 18

In the high density Spent Fuel Rack design (Ref. 9), the spent fuel storage pool is divided into four separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel which has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

Region 1 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 +/- 0.05 wt% U-235, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type A fuel assemblies with enrichment, burnup, and cooling times in accordance with Design Feature 5.6.1.1.c.1. Region 2 is designed to accommodate Type A or Type T fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment burned to an assembly average burnup of at least 30.27 MWD/kgU for Type A fuel or 33.1095 MWD/kgU for Type T fuel, or other enrichment with a burnup yielding an equivalent reactivity in the fuel racks in accordance with Design Feature 5.6.1.1.c.2. Region 3 is designed to accommodate fresh fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or fuel assemblies of any lower reactivity in a 2-of-4 checkerboard arrangement with water-filled cells in accordance with Design Feature 5.6.1.1.c.3. Region 4 is designed to accommodate fresh fuel up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type T fuel assemblies with burnup and cooling times in accordance with Design Feature 5.6.1.1.c.4.

BACKGROUND (continued)

severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks may therefore be achieved by controlling the location of each assembly in accordance with Design Features 5.6 FUEL STORAGE. During fuel movement, it is necessary to perform Surveillance Requirement 4.7.13.2.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of any one of the three regions. Examples of these accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in all regions. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ± 0.05 w/o U-235 ***fresh*** fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement. The amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 due to this fuel misload accident is 700 ppm (Ref. 1 ***and Ref. 9***).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the k_{eff} design basis limit of 0.95 (Ref. 6). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event.

The concentration of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The spent fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the spent fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 7. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel storage pool.

(continued)

PLANT SYSTEMS

BASES (continued)

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

Action a:

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored along with suspending movement of fuel assemblies.

Action a is modified by a provision indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4 is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

Surveillance 4.7.13.1

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no significant replenishment of pool water is expected to take place over such a short period of time. (Ref. 6)

Surveillance 4.7.13.2

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit during fuel movement until the final configuration of the assemblies in the storage racks is verified to be correct. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 72 hour Frequency provides additional assurance that the maximum k_{eff} remains below the 0.95 limit under the postulated accident condition. (Ref. ~~8~~ ***1, 8 and 9***)

(continued)

PLANT SYSTEMS

BASES (continued)

REFERENCES

1. Stanley E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349

2. B.K. Grimes (NRC GL78011), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978

3. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998

4. UFSAR, Section 4.3.2.7, "Critically of Fuel Assemblies"

5. 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).

6. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302

7. FSAR, Section 15.4.5

8. NRC letter to TVA dated August 1, 1990, " Increase Fuel Enrichment to 5.0 Weight Percent (TAC Nos. 76074, 76075, 76774, 76775) (TS 90-12) - Sequoyah Nuclear Plant, Units 1 and 2"

9. ***Stanley E. Turner (Holtec International), "Evaluation of the Effect of the Use of Tritium Producing Burnable Absorber Rods (TPBARs) on Fuel Storage Requirements", HI-2012629***

PLANT SYSTEMS **[This page deleted]**

BASES

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

BACKGROUND

The Sequoyah cask pit pool consists of a deep pool with adjacent shelf area. The cask pit pool is connected to the spent fuel pool through a weir gate. The cask pit is intended to be used for spent fuel shipment activities.

High density spent fuel storage racks have been approved for addition and use in the cask loading area of the cask pit (Ref. 1) but presently are not installed. The 15 x 15 module could store 225 fuel assemblies and is designed to maintain stored fuel having an initial enrichment of up to 5 wt % U-235, in a safe, coolable, and sub-critical configuration during normal discharge, full core offload storages and postulated accident conditions. Fuel assemblies shall be stored in accordance with paragraph 5.6.1.1.d in Design Features 5.6, Fuel Storage.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of the cask pit. Examples of accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the cask pit pool prevents criticality. The most limiting postulated accident bounding the cask pit pool has been determined to occur in the spent fuel pool. The postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 $\pm$ 0.05 w/o U-235 fuel assembly into an storage cell location in the Region 2 checkerboard storage arrangement for an irradiated fuel assembly. The amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 due to this fuel misload accident is 700 ppm (Ref. 2).

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The cask pit pool boron concentration is required to be $\geq$ 2000 ppm. The specified concentration of dissolved boron in the cask pit pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the cask pit pool.

(continued)

PLANT SYSTEMS

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BASES (continued)

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the cask pit pool.

ACTIONS

Action a:

When the concentration of boron in the cask pit pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored along with suspending movement of fuel assemblies.

Action a is modified by a provision indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4 is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

Surveillance 4.7.14.1

This Surveillance Requirement verifies that the concentration of boron in the cask pit pool is within the required limit. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no significant replenishment of pool water is expected to take place over such a short period of time. (Ref. 4)

Surveillance 4.7.14.2

This Surveillance Requirement verifies that the concentration of boron in the cask pit pool is within the required limit during fuel movement until the final configuration of the assemblies in the storage racks is verified to be correct. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 72 hour Frequency provides additional assurance that the maximum k_{eff} remains below the 0.95 limit under the postulated accident condition. (Ref. 1)

(continued)

PLANT SYSTEMS **[This page deleted]**

BASES (continued)

REFERENCES

1. NRC letter to TVA dated April 28, 1993, "Issuance of Amendments (TAC Nos. M83068 and M83069)"
2. Stanley E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349
3. FSAR, Section 15.4.5
4. K. K. Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

INDEX

DESIGN FEATURES

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Source Range Nuclear Flux	NOTE 1	~~1 to 1 x 10^6~~ cps ⇑ ***0.1 to 1 x 10^5***	1
2. Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3. Reactor Coolant Temperature - Hot Leg	NOTE 1	0-650°F	1/loop
4. Pressurizer Pressure	NOTE 1	0-3000 psig	1
5. Pressurizer Level	NOTE 1	0-100%	1
6. Steam Generator Pressure	NOTE 1	0-1200 psig	1/steam generator
7. Steam Generator Level	NOTE 2 or near Auxilary F. W. Pump	0-100%	1/steam generator
8. Deleted			
9. RHR Flow Rate	NOTE 1	0-4500 gpm	1
10. RHR Temperature	NOTE 1	50-400°F	1
11. Auxiliary Feedwater Flow Rate	NOTE 1	0-440 gpm	1/steam generator

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7615 and 7960 gallons of borated water,

2

c. Between ~~2400~~ 3500⇑ and ~~2700~~ 3800⇑ ppm of boron,

d. A nitrogen cover-pressure of between 624 and 668 psig, and

e. Power removed from isolation valve when RCS pressure is above 2000 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

* Pressurizer pressure above 1000 psig.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between ~~2500~~ 3600⇑ and ~~2700~~ 3800⇑ ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

[This page deleted]

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The cask pit pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: Whenever fuel assemblies are stored in the cask pit rack.

ACTION:

a. With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and initiate action to restore cask pit pool boron concentration to within limit. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.1 Verify at least once per 7 days the cask pit pool boron concentration is within limit.

4.7.14.2 Verify at least once per 72 hours during fuel movement the cask pit pool boron concentration is within limit and until the configuration of the assemblies in the storage rack is verified to comply with the criticality loading criteria specified in Design Feature 5.6.1.1.d.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-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. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor, as described in the Framatome Cogema Fuels Report BAW-2328, beginning with the Unit 2 Operating Cycle 10 core.

INSERT

Sequoyah is authorized to place a maximum of 2256 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

~~CRITICALITY~~ - SPENT FUEL
[Insert 1]

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

a. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron.*

b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Arrangements of one or more of three different arrays (Regions) or sub-arrays as illustrated in Figures 5.6-1 and 5.6-1a. These arrangements in the spent fuel storage pool have the following definitions:

Type A

1. Region 1 is designed to accommodate new fuel ~~with a~~ maximum enrichment ~~of~~ 4.95 ± 0.05 wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh ~~assembly~~ with 3 spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 5.6-2 and defined by the ~~equations in Table 5.6-1. Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.~~ The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-2 or Table 5.6-1.

   Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or integral fuel burnable absorber (IFBA). For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 5.6-2.

Restrictions in Region 1

Any of the three sub-arrays illustrated in Figure 5.6-1a may be used in any combination provided that:

~~1~~ ***A)*** Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-1 and 5.6-2, as appropriate.

~~2~~ ***B)*** The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

~~3~~ ***C)*** If Region 1 arrays are used in conjunction with ~~Region 2~~ ***3*** ⇑ or Region ~~3~~ ***4*** ⇑ arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 5.6-1).

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

Insert 1

For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel that has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

5.6 FUEL STORAGE

[Insert 2]

2. Region 2 is designed to accommodate ***[Insert 3]*** fuel of 4.95±0.05 wt% U-235 initial enrichment burned to at least 30.27 ***[Insert 4]*** 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 in MWD/KgU and cooling time is given by the equations in Table 5.6-3 ***[Insert 5]*** ~~in terms of E, where E is the initial enrichment in the axial zone of highest enrichment (wt% U-235)~~. The minimum required burnups are illustrated in Figure 5.6-3 ***[Insert 5]*** in terms of the initial enrichment and cooling time.

Restrictions in Region 2

The following restrictions apply to the storage of spent fuel in the Region 2 cells:

~~1~~ ***A***) The spent fuel shall conform to the minimum burnup requirements defined by the equations in Table 5.6-3 ***[Insert 6]***. Linear interpolation between cooling times may be made if desired.

~~2~~ ***B***) For the interface with Region 1 ***[Insert 7]***storage cells, fresh fuel in Region 1 ***[Insert 7]***shall not be stored adjacent to spent fuel assemblies in the Region 2 storage cells.

[Insert 8]

3. Region 3 is designed to accommodate fuel of 4.95±0.05 wt% U-235 initial enrichment (or fuel assemblies of any lower reactivity) in a 2-out-of-4 checkerboard arrangement with water-filled cells. The water-filled cells shall not contain any components bearing any fissile material, but may accommodate miscellaneous items or equipment.

Restrictions in Region 3

[Insert 9]

~~1~~ ***A***) For the interface between Region ~~1~~ ⇑ ***3*** and Region ~~3~~ ⇑ ***1 or 4*** storage regions, fresh fuel assemblies shall not be stored adjacent to each other.

~~2~~ ***B***) If miscellaneous ⇑ ***non-fissile bearing*** items or equipment are stored in the water cells of Region 3, the total volume of the miscellaneous items shall be no more than 75% of the storage cell volume.

~~3~~ ***C***) No ⇑ ***loose*** fuel rods~~, assemblies,~~ or items containing fissile material shall be stored in the water cells of Region 3.

[Insert 10]

~~An empty cell is less reactive than any cell containing fuel and therefore may be used as a Region 1, Region 2, or Region 3 cell in any arrangement.~~

~~d. Region 2 array described above may be used in the 15 x 15 storage rack module in the cask loading area of the cask pit.~~

[Insert 11]

e. A nominal concentration of 2000 ppm boron ~~is~~ ***is*** in the pool water. This concentration of soluble boron provides a margin sufficient to allow timely detection of a boron dilution accident and corrective action before the minimum concentration (700 ppm) required to protect against the most severe postulated fuel handling accident or before the minimum concentration (300 ppm) required to maintain the storage configuration design basis (k_{eff} less than 0.95) is reached.

Insert 2

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 1, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

Insert 3

Type A or Type T

Insert 4

(Type A) or 33.1095 (Type T)

Insert 5

(Type A) or 5.6-4 (Type T)

Insert 6

or 5.6-4, as appropriate

Insert 7

or 4

Insert 8

C) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 2, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

Insert 9

The following restrictions apply to the storage of fuel in the Region 3 cells:

Insert 10

4. Region 4 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235 (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with three Type T spent fuel assemblies having burnup and cooling times illustrated in Figure 5.6-5 and defined by the equations in Table 5.6-5. The presence of either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-5 or Table 5.6-5.

Insert 10 continued

One alternative storage array (or sub-array) is acceptable in Region 4 if the fresh fuel contains rods with gadolinia fuel burnable absorber. For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array is defined by the equations in Table 5.6-6 and illustrated in Figure 5.6-6. For fresh assemblies containing more than eight (8) gadolinia bearing fuel rods, the limiting burnup for eight (8) gadolinia rods shall apply.

Restrictions in Region 4

Any of the two sub-arrays illustrated in Figure 5.6-1a applying to Region 4 storage may be used in any combination provided that:

A) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-5 and 5.6-6, as appropriate.

B) The arrangement of Region 4 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

C) If Region 4 arrays are used in conjunction with Region 1 or 3 arrangements, the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see Figure 5.6-1)

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 4, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

Insert 11

d. An empty cell (or a cell containing non-fissile bearing miscellaneous items displacing no more than 75% of the storage cell volume) is less reactive than any cell containing fuel and therefore may be used as a Region 1, 2, 3, or 4 cell in any arrangement.

5.6 FUEL STORAGE

CRITICALITY - NEW FUEL

5.6-7

5.6.1.2 The new fuel pit storage racks are designed for fuel enriched to 5.0 weight percent U-235 and shall be maintained with the arrangement of 146 storage locations shown in Figure ~~5.6-4~~. The cells shown as empty cells in Figure ~~5.6-4~~ shall ~~have~~ physical barriers installed to ensure that inadvertant loading of fuel assemblies into these locations does not occur. This configuration ensures k_{eff} will remain less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 under optimum moderation conditions.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

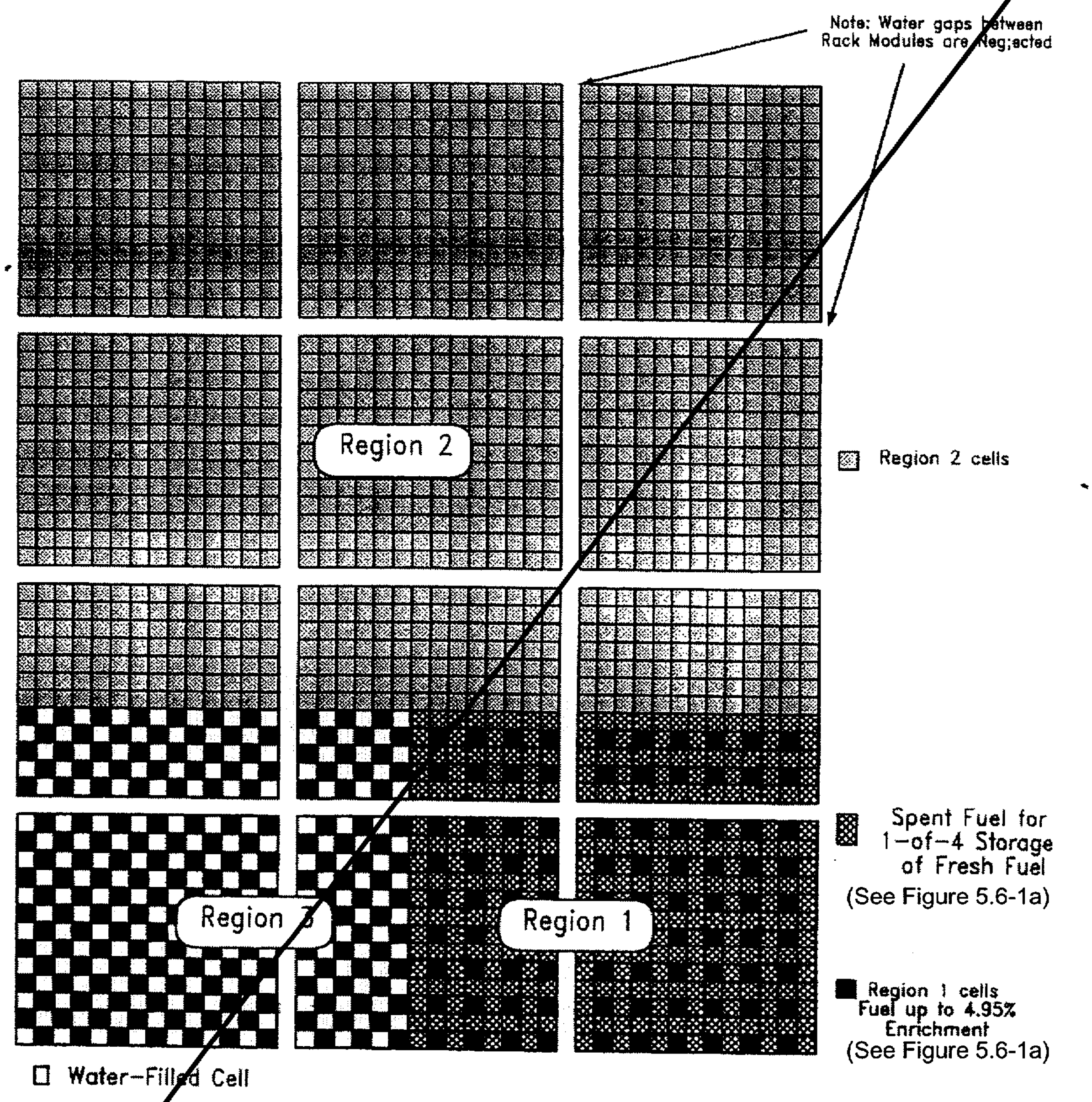
CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. ~~In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.~~

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

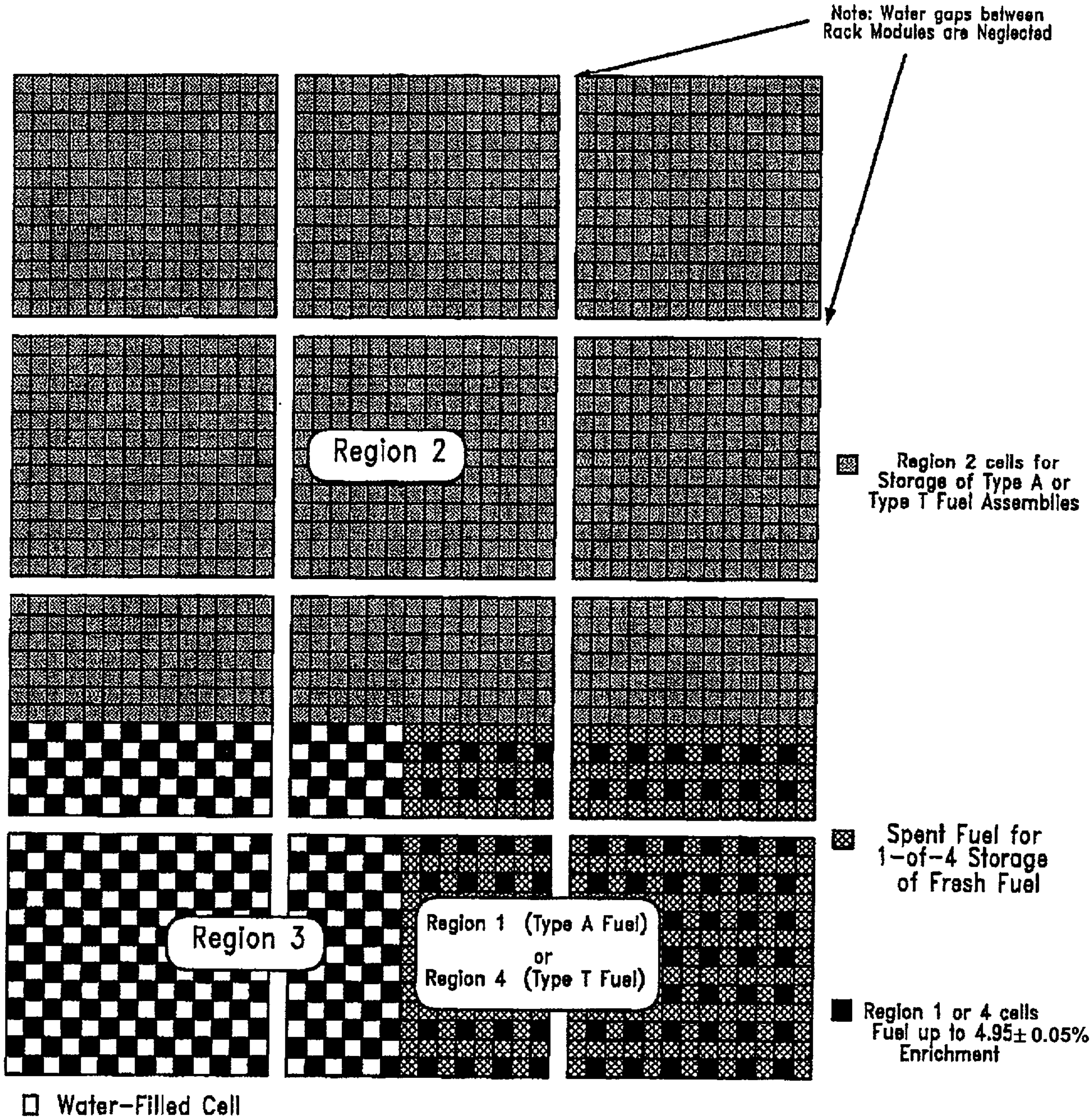
[Replace with Insert 12]



Note: The edges of the sketch above are not necessarily the edges of the pool. The Regions may appear anywhere in the pool and in any orientation, subject to the restriction in Design Feature 5.6.1.1.c.

Figure 5.6-1
Arrangements of Fuel Storage Regions in the Sequoyah Spent Fuel Storage Pool

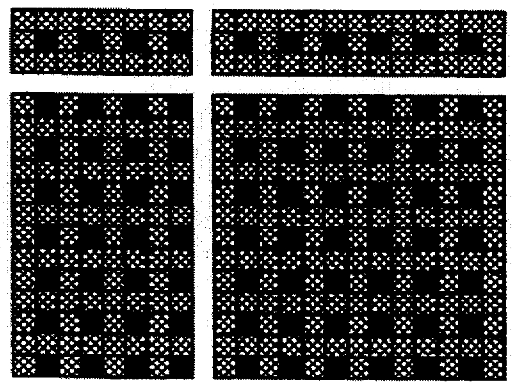
Insert 12



Note: The edges of the sketch above are not necessarily the edges of the pool. The Regions may appear anywhere in the pool and in any orientation, subject to the restrictions in Design Features 5.6.1.1.c.

FIG 5.6-1 Arrangements of Fuel Storage Regions in the Sequoyah Spent Fuel Storage Pool

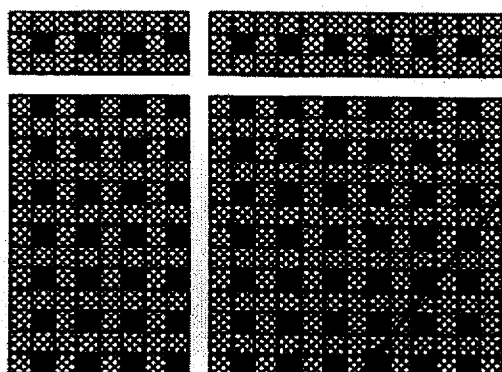
[Replace with Insert 13]



Spent Fuel for Region 1 Storage 1-of-4 Pattern

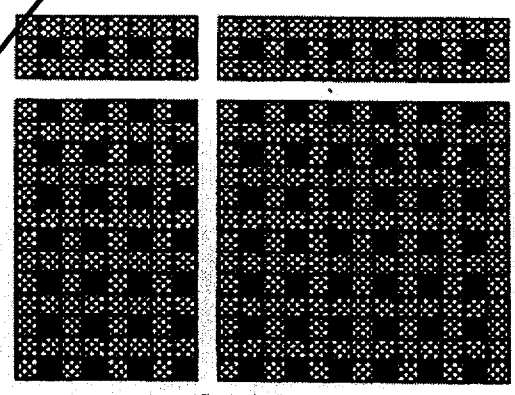
Fresh Fuel for Region 1 cells No Gd or IFBA

NOTE: WHEN CREDIT IS TAKEN FOR GADOLINIA OR IFBA RODS IN FRESH ASSEMBLIES THE SPENT FUEL ASSEMBLIES NEED NOT HAVE CONTAINED GADOLINIA OR IFBA RODS..



Spent Fuel for Region 1 Storage 1-of-4 Pattern

Fresh Fuel for Region 1 cells With Gadolinia

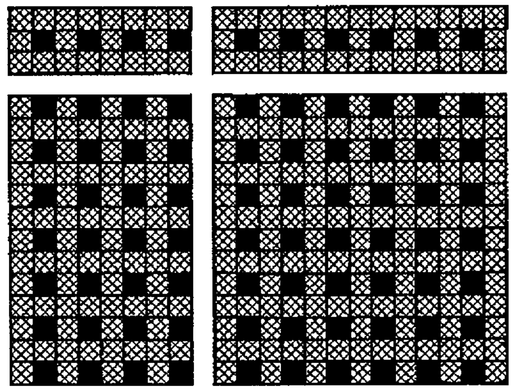


Spent Fuel for Region 1 Storage 1-of-4 Pattern

Fresh Fuel for Region 1 cells with IFBA Rods

Figure 5.6-1a
Acceptable Spent Fuel Pool Loading Patterns for Checkerboard Storage of Fresh and Spent Fuel Assemblies - Example

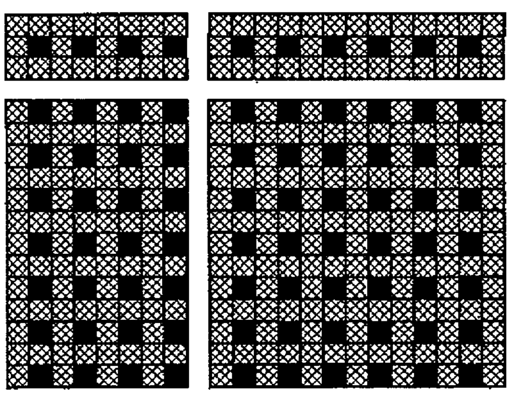
Insert 13



Spent Fuel for Storage in Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

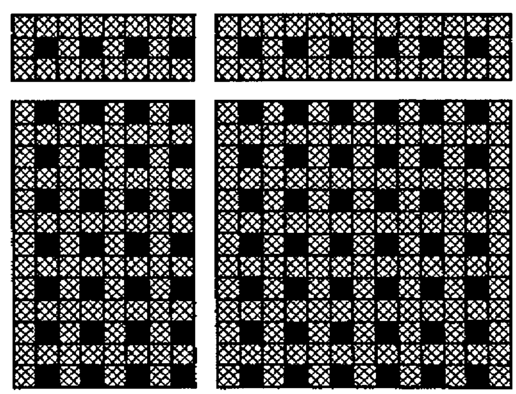
Fresh Fuel for Region 1 or Region 4 cells (No Gd or IFBA)

NOTE: WHEN CREDIT IS TAKEN FOR GADOLINIA RODS IN FRESH ASSEMBLIES THE SPENT FUEL ASSEMBLIES NEED NOT HAVE CONTAINED GADOLINIA RODS..



Spent Fuel for Storage of Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

Fresh Fuel with Gadolinia for Region 1 or Region 4 cells



Spent Fuel for Storage in Region 1 in 1-of-4 Pattern

Fresh Fuel with IFBA Rods for Storage in Region 1 only (1-of-4 Pattern)

Fig. 5.6-1a Acceptable Storage Patterns for Checkerboard Storage of Fresh and Spent Fuel Assemblies in Region 1 or Region 4 – Example

for Type A Fuel

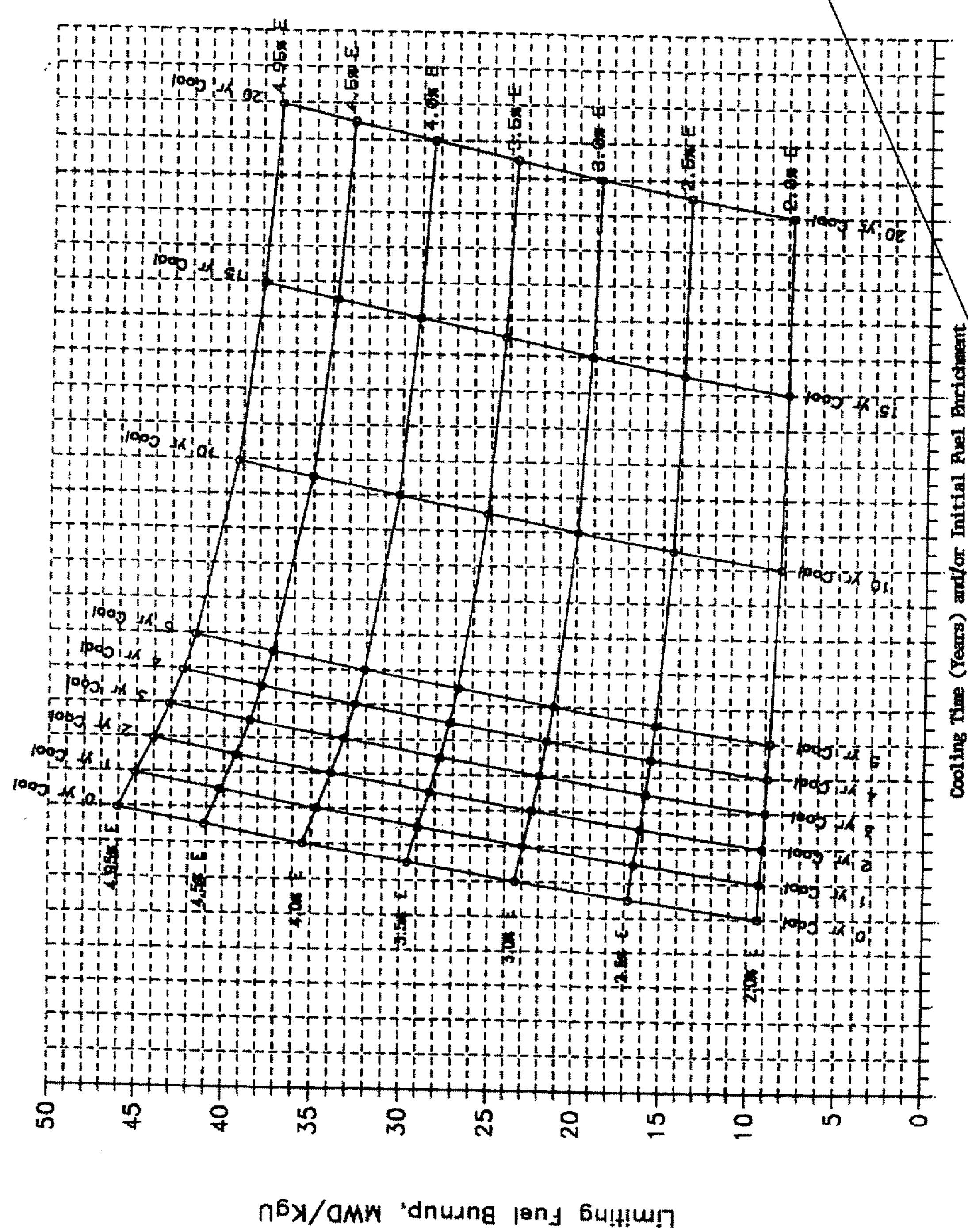


Fig. 5.6-2 3-Dimensional Plot of Minimum Fuel Burnups in Region 1 for Enrichments and/or Cooling Times

For Type A Fuel

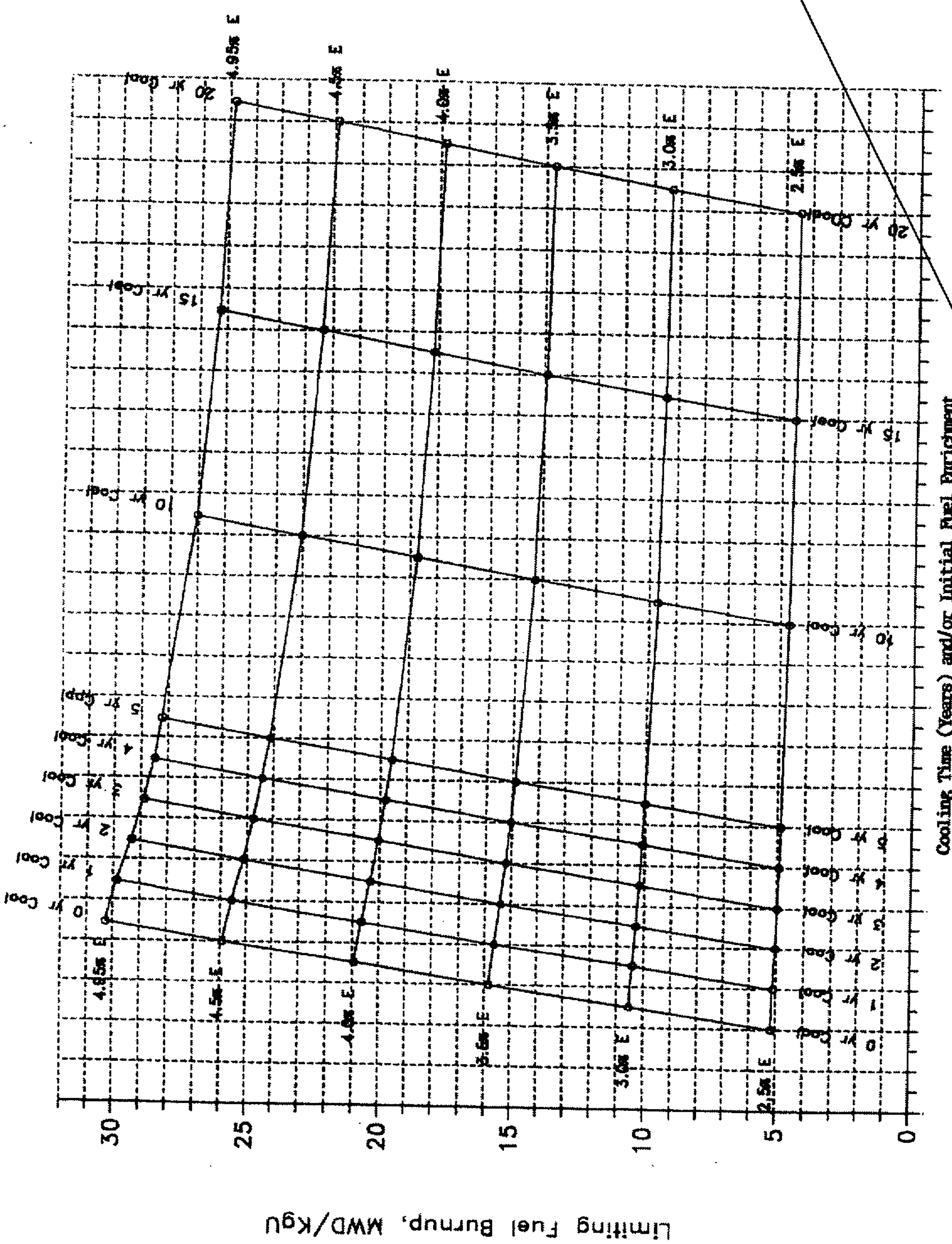
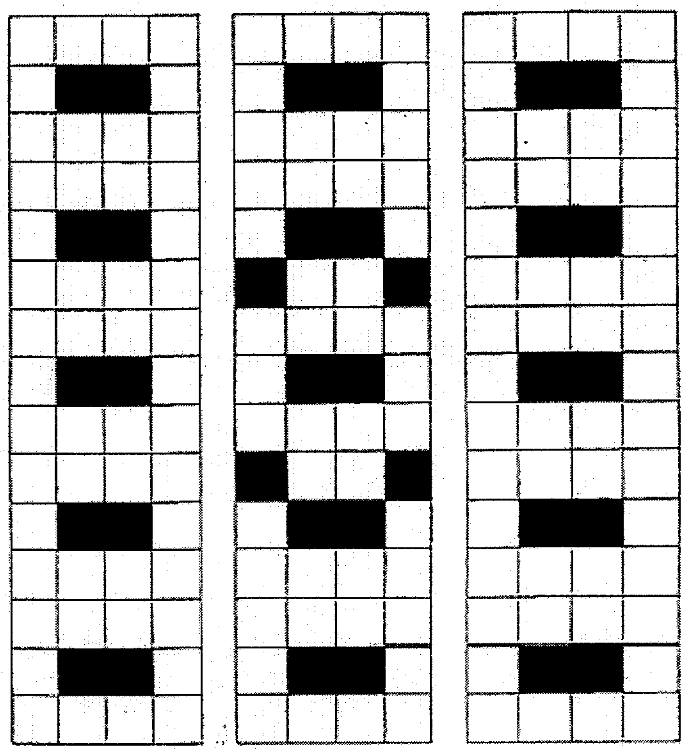


Fig. 5.6-3 3- Dimensional Plot of Minimum Fuel Burnups in Region 2 for Enrichments and Cooling Times

[Add Inserts 14, 15, and 16]



Basic Cell 21 inch X 21 inch

Empty Cell

9 - 4 X 5 Cell Racks

146 / 180 Loading Pattern

Figure ~~5.6-4~~ ***5.6-7***
New Fuel Pit Storage Rack Loading Pattern

Insert 14

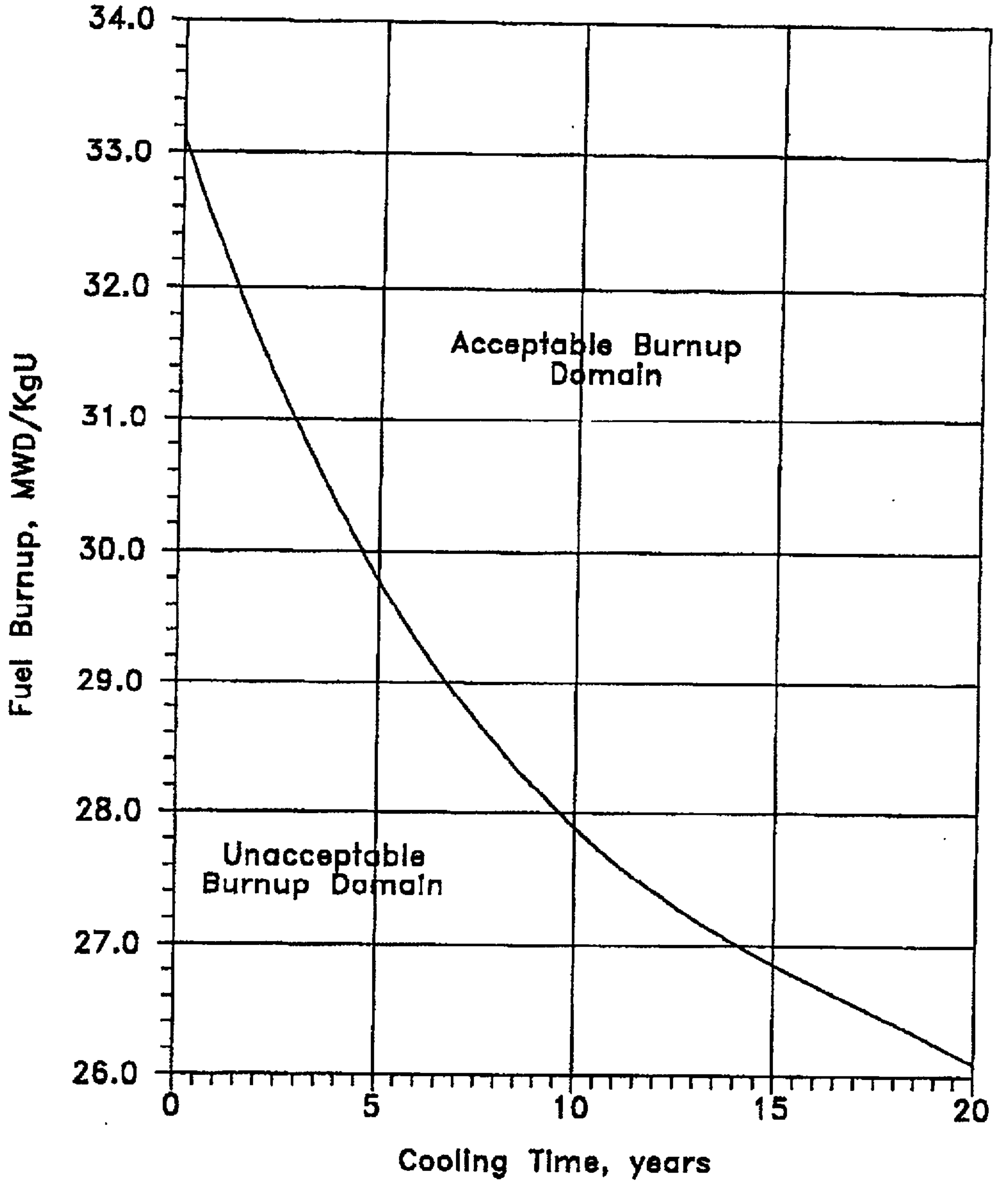


Fig. 5.6-4 Limiting Burnup Requirements in Region 2 for Face Adjacent Storage of Type T Spent Fuel Assemblies

Insert 15

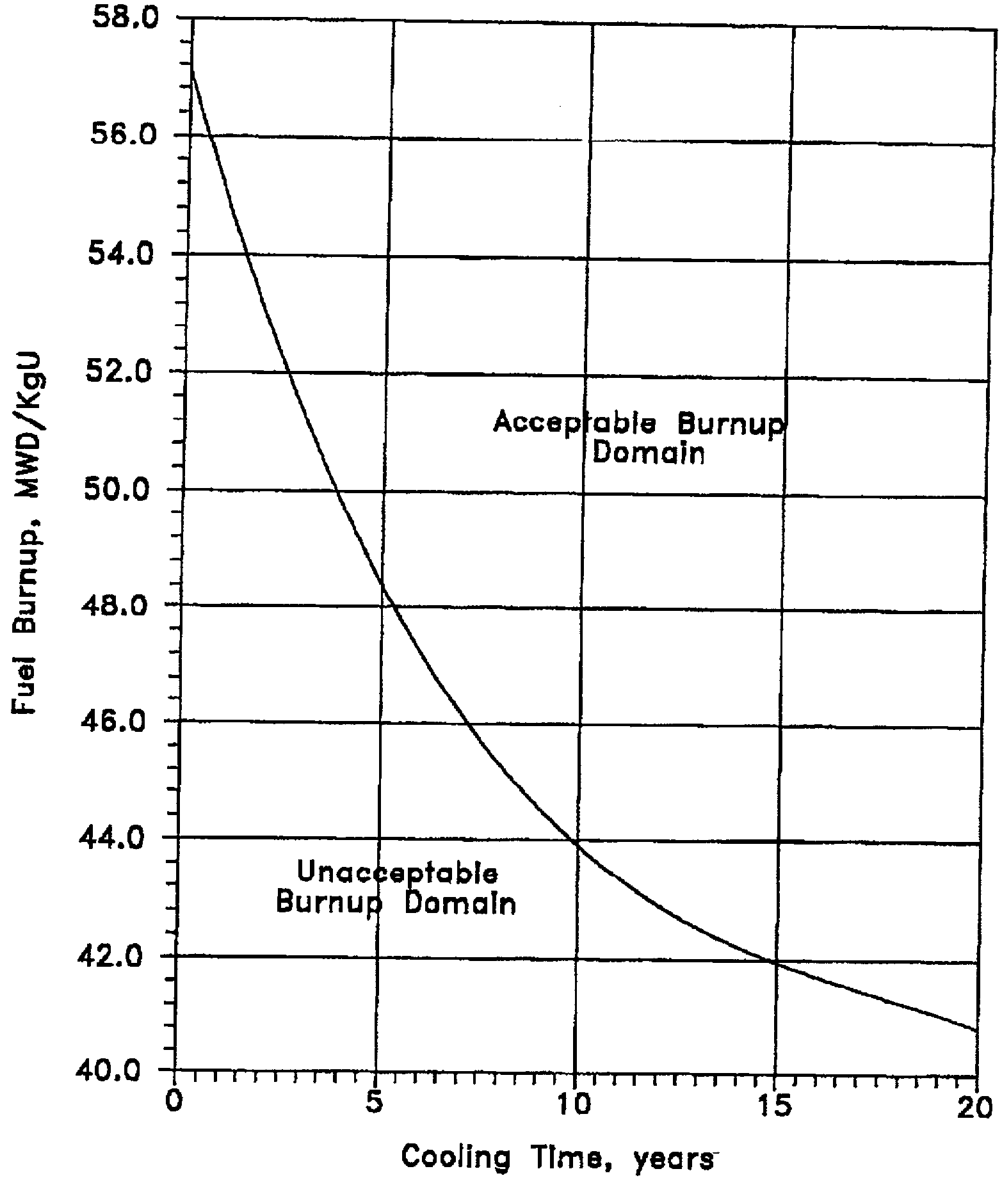


Fig. 5.6-5 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh and 3 Type T Spent Fuel Assemblies

Insert 16

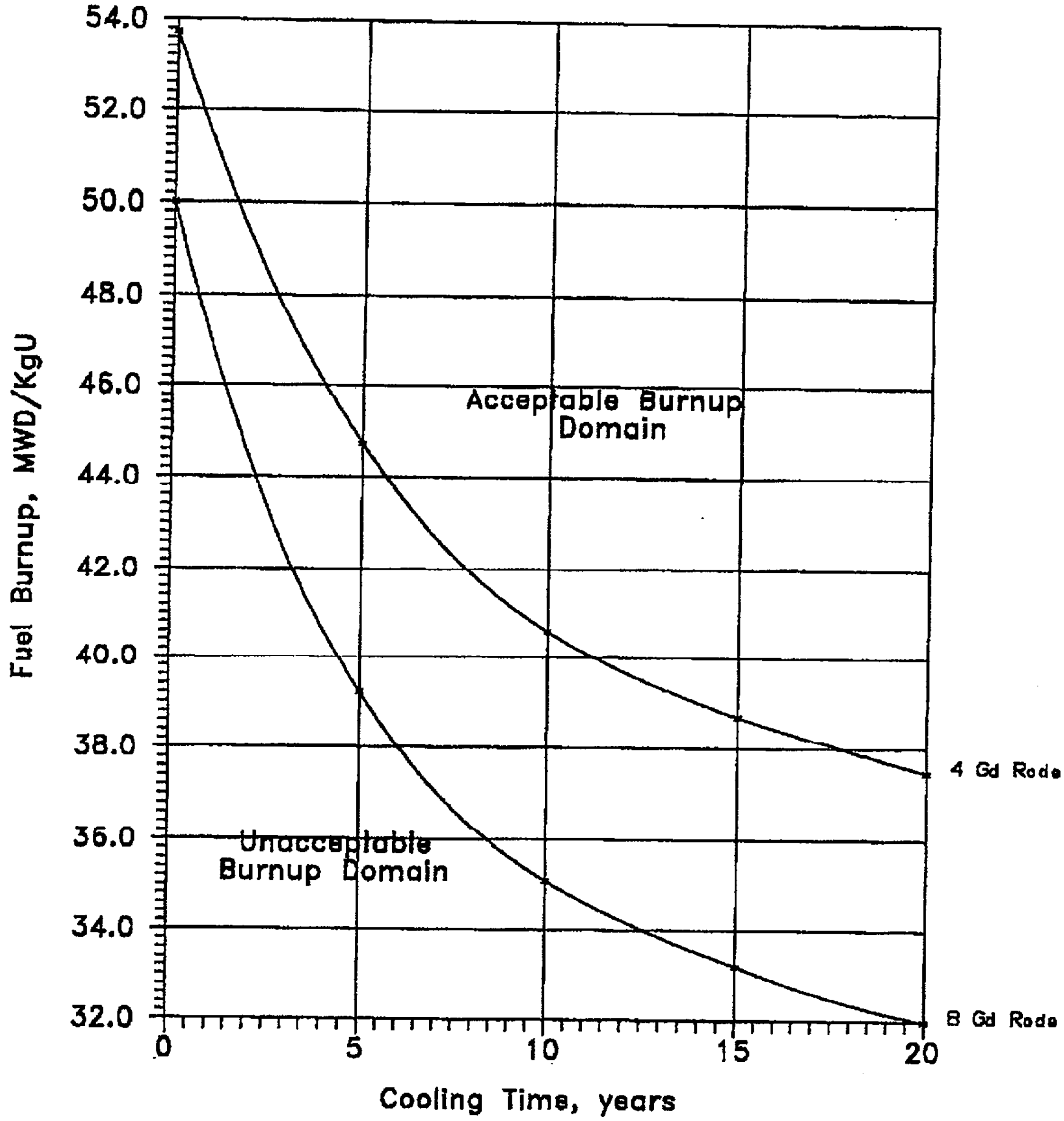


Fig. 5.6-6 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh (with Gadolinia) and 3 Type T Spent Fuel Assemblies

Table 5.6-1
Region 1 Storage Burnup Restrictions: Checkerboard of 1 Fresh Fuel Assembly ⇓ and 3 ***Type A*** Spent Fuel Assemblies ~~(Without Gadolinium or IFBA Rods)~~ ***(Without Gadolinium or IFBA Rods)***

For Zero Year Cooling Time Bu (limit) = - 28.1868 + 23.0765 x E – 2.46264 x E^2 + 0.167868 x E^3
For One Year Cooling Time Bu (limit) = - 27.3317 + 22.5087 x E – 2.40586 x E^2 + 0.164207 x E^3
For Two Years Cooling Time Bu (limit) = -26.4693 + 21.8404 x E – 2.31873 x E^2 + 0.158218 x E^3
For Three Years Cooling Time Bu (limit) = -25.7404 + 21.2659 x E – 2.24287 x E^2 + 0.153018 x E^3
For Four Years Cooling Time Bu (limit) = - 25.1367 + 20.7910 x E –2.18484 x E^2 + 0.1499363 x E^3
For Five Years Cooling Time Bu (limit) = - 24.5981 + 20.3568 x E – 2.12719 x E^2 + 0.145431 x E^3
For Ten Years Cooling Time Bu (limit) = - 23.2050 + 19.2969 x E – 2.06993 x E^2 + 0.145875 x E^3
For Fifteen Years Cooling Time Bu (limit) = -22.6098 + 18.8544 x E – 2.08617 x E^2 + 0.150473 x E^3
For Twenty Years Cooling Time Bu (limit) = - 22.3017 + 18.622 x E – 2.11206 x E^2 + 0.15467 x E^3

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-2
Region 1 Storage Burnup Restrictions with Gadolinium or IFBA ***in Fresh Fuel***

With Gadolinium Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 ***Type A*** Spent Fuel Assemblies

Zero Year Cooling Time, 0 Gadolinia Rods Bu (limit) = $-28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 4 Gadolinia Rods Bu (limit) = $-28.4012 + 22.0062 \times E - 2.19268 \times E^2 + 0.143601 \times E^3$
Zero Year Cooling Time, 8 Gadolinia Rods Bu (limit) = $-31.4262 + 22.0768 \times E - 2.38845 \times E^2 + 0.164888 \times E^3$

Note: If more that 8 Gadolinium rods per assembly, use the 8 rod correlation

With IFBA Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 ***Type A*** Spent Fuel Assemblies

Zero Year Cooling Time, 0 IFBA Rods Bu (limit) = $-28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 16 IFBA Rods Bu (limit) = $-28.5048 + 21.6411 \times E - 2.15262 \times E^2 + 0.140904 \times E^3$
Zero Year Cooling Time, 32 IFBA Rods Bu (limit) = $-31.0949 + 22.0435 \times E - 2.36088 \times E^2 + 0.162229 \times E^3$
Zero Year Cooling Time, 48 IFBA Rods Bu (limit) = $-33.1342 + 22.3999 \times E - 2.55367 \times E^2 + 0.18082 \times E^3$
Zero Year Cooling Time, 64 IFBA Rods Bu (limit) = $-36.0468 + 24.1492 \times E - 3.11807 \times E^2 + 0.233987 \times E^3$

Note: If more that 64 IFBA rods per assembly, use the correlation for 64 IFBA rods
Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-3
Region 2 ~~Storage Burnup Restrictions~~
For Type A Fuel

Zero Cooling Time Bu (limit) = - 23.8702 + 12.3026 x E - 0.275672 x E^2
1 Year Cooling Time Bu (limit) = - 23.6854 + 12.2384 x E - 0.287498 x E^2
2 Years Cooling Time Bu (limit) = - 23.499 + 12.1873 x E - 0.305988 x E^2
3 Years Cooling Time Bu (limit) = - 23.3124 + 12.1249 x E - 0.319566 x E^2
4 Years Cooling Time Bu (limit) = - 23.1589 + 12.0748 x E - 0.332212 x E^2
5 Years Cooling Time Bu (limit) = - 22.6375 + 11.7906 x E - 0.307623 x E^2
10 Years Cooling Time Bu (limit) = - 21.7256 + 11.3660 x E – 0.31029 x E^2
15 Years Cooling Time Bu (limit) = - 21.1160 + 11.0663 x E - 0.306231 x E^2
20 Years Cooling Time Bu (limit) = - 20.6055 + 10.7906 x E - 0.29291 x E^2

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-4

Face Adjacent Storage of Type T Spent Fuel (Region 2)

$$Bu\ (limit) = 33.1095 - 0.845146 \times CT + 0.0399888 \times CT^2 - 0.000762846 \times CT^3$$

Table 5.6-5

Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel (Region 4: 1 Fresh Assembly and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

$$Bu\ (limit) = 57.118 - 2.13277 \times CT + 0.0772537 \times CT^2 + 0.00127446 \times CT^3 - 9.15855\ E\text{-}5 \times CT^4$$

Table 5.6-6

Gadolinia Credit: Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel (Region 4: 1 Fresh Assembly with Gadolinia and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

4 Gadolinia Rods

$$Bu\ (limit) = 53.73 - 2.5265 \times CT + 0.172283 \times CT^2 - 0.00585995 \times CT^3 + 0.0000766655 \times CT^4$$

8 Gadolinia Rods

$$Bu\ (limit) = 50.00 - 3.26817 \times CT + 0.276117 \times CT^2 - 0.0117934 \times CT^3 + 0.000195334 \times CT^4$$

Note: 1. If more than 8 gadolinia rods per assembly, use the 8 rod correlation

2. BU = Fuel Burnup, MWD/Kg-U; CT = Cooling Time of Spent Fuel Assemblies, Years

, and 4) tritium and hydrogen that exist in the Tritium Producing Burnable Absorber Rods prior to the accident

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the hydrogen mitigation system, consisting of 68 hydrogen igniters per unit, is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, ~~and~~ 3) corrosion of metals within containment. These hydrogen control systems are designed to mitigate the effects of an accident as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," Revision 2, dated November 1978. The hydrogen monitors of Specification 3.6.4.1 are part of the accident monitoring instrumentation in Specification 3.3.3.7 and are designated as Type A, Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The operability of at least 66 of 68 igniters in the hydrogen control distributed ignition system will maintain an effective coverage throughout the containment. This system of ignitors will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1071 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,082,024 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the

PLANT SYSTEMS

BASES

Reports HI-992349 (Ref 1) and HI-2012629 (Ref 9)

3/4.7.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION

BACKGROUND The spent fuel racks have been analyzed in accordance with the Holtec International methodology contained in Holtec ~~Report HI - 992349 (Ref. 1)~~. This methodology ensures that the spent fuel rack multiplication ~~factor, k_{eff}~~ is less than or equal to 0.95, as recommended by the NRC guidance contained in NRC Letter to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 and USNRC Internal Memorandum from L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998 (Refs. 2 & 3). The codes, methods, and techniques contained in the methodology are used to satisfy the k_{eff} criterion. The spent fuel storage racks were analyzed using Westinghouse 17x17 V5H fuel assemblies, with enrichments up to 4.95 ±0.05 w/o U-235 and configurations which take credit for checkerboarding, burnup, soluble boron, integral fuel burnable absorbers (such as IFBA or gadolinia), and cooling time to ensure that k_{eff} is ~~maintained ≤ 0.95,~~ including uncertainties, tolerances, and accident conditions. ***[Insert 17]*** In addition, the SFP k_{eff} is maintained < 1.0, including uncertainties, ~~tolerances~~ on a 95/95 basis without any soluble boron. Calculations were performed to evaluate the reactivity of fuel types used at SQN. The results show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity, thereby bounding all fuel types utilized and ~~stored at SQN.~~

Replace with Insert 18 ⇒

~~In the high density Spent Fuel Storage Rack design (Refs. 1 and 4), the spent fuel storage pool is divided into three separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, or spent fuel regardless of the discharge fuel burnup in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 spent fuel assemblies with enrichment, burnup and cooling times in accordance with Design Features 5.6.1.1.c.1. Region 2 is designed to accommodate fuel which have 4.95 ± 0.05 wt% U-235 initial enrichment burned to at least 30.27 MWD/KgU (assembly average), or fuel of other enrichment with a burnup yielding an equivalent reactivity in the fuel racks in accordance with Design Features 5.6.1.1.c.2. Region 3 is designed to accommodate fuel of 4.95 ± 0.05 wt% U-235 initial enrichment or fuel assemblies of any lower reactivity in a 2-out-of-4 checkerboard arrangement with water-filled cells and in accordance with Design Features 5.6.1.1.c.3.~~

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 < 1.0 be evaluated in the absence of soluble boron. Hence, the design of all 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 (Ref. 5) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most

(continued)

Insert 17

The analysis also accounts for the reactivity effects of operating the fuel with discrete burnable poisons (such as burnable poison rod absorbers or tritium producing burnable absorber rods).

Insert 18

In the high density Spent Fuel Rack design (Ref. 9), the spent fuel storage pool is divided into four separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel which has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

Region 1 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 +/- 0.05 wt% U-235, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type A fuel assemblies with enrichment, burnup, and cooling times in accordance with Design Feature 5.6.1.1.c.1. Region 2 is designed to accommodate Type A or Type T fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment burned to an assembly average burnup of at least 30.27 MWD/kgU for Type A fuel or 33.1095 MWD/kgU for Type T fuel, or other enrichment with a burnup yielding an equivalent reactivity in the fuel racks in accordance with Design Feature 5.6.1.1.c.2. Region 3 is designed to accommodate fresh fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or fuel assemblies of any lower reactivity in a 2-of-4 checkerboard arrangement with water-filled cells in accordance with Design Feature 5.6.1.1.c.3. Region 4 is designed to accommodate fresh fuel up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type T fuel assemblies with burnup and cooling times in accordance with Design Feature 5.6.1.1.c.4.

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BASES

BACKGROUND (continued)

severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks may therefore be achieved by controlling the location of each assembly in accordance with Design Features 5.6 FUEL STORAGE. During fuel movement, it is necessary to perform Surveillance Requirement 4.7.13.2.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of any one of the three regions. Examples of these accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in all regions. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ± 0.05 w/o U-235 ***fresh*** fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement. The amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 due to this fuel misload accident is 700 ppm (Ref. 1 ***and Ref. 9***).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the k_{eff} design basis limit of 0.95 (Ref. 6). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event.

The concentration of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The spent fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the spent fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 7. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel storage pool.

(continued)

PLANT SYSTEMS

BASES (continued)

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

Action a:

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored along with suspending movement of fuel assemblies.

Action a is modified by a provision indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4 is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

Surveillance 4.7.13.1

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no significant replenishment of pool water is expected to take place over such a short period of time. (Ref. 6)

Surveillance 4.7.13.2

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit during fuel movement until the final configuration of the assemblies in the storage racks is verified to be correct. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 72 hour Frequency provides additional assurance that the maximum ~~k_{eff} remains below the~~ 0.95 limit under the postulated accident condition. (Ref. ~~8~~ ***1, 8, and 9***)

(continued)

PLANT SYSTEMS

BASES (continued)

REFERENCES

1. Stanley E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349

2. B.K. Grimes (NRC GL78011), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978

3. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998

4. UFSAR, Section 4.3.2.7, "Critically of Fuel Assemblies"

5. 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).

6. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302

7. FSAR, Section 15.4.5

8. NRC letter to TVA dated August 1, 1990, " Increase Fuel Enrichment to 5.0 Weight Percent (TAC Nos. 76074, 76075, 76774, 76775) (TS 90-12) - Sequoyah Nuclear Plant, Units 1 and 2"

9. ***Stanley E. Turner (Holtec International), "Evaluation of the Effect of the Use of Tritium Producing Burnable Absorber Rods (TPBARs) on Fuel Storage Requirements", HI-2012629***

PLANT SYSTEMS **[This page deleted]**

BASES

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

BACKGROUND

The Sequoyah cask pit pool consists of a deep pool with adjacent shelf area. The cask pit pool is connected to the spent fuel pool through a weir gate. The cask pit is intended to be used for spent fuel shipment activities.

High density spent fuel storage racks have been approved for addition and use in the cask loading area of the cask pit (Ref. 1) but presently are not installed. The 15 x 15 module could store 225 fuel assemblies and is designed to maintain stored fuel having an initial enrichment of up to 5 wt % U-235, in a safe, coolable, and sub-critical configuration during normal discharge, full core offload storages and postulated accident conditions. Fuel assemblies shall be stored in accordance with paragraph 5.6.1.1.d in Design Features 5.6, Fuel Storage.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of the cask pit. Examples of accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the cask pit pool prevents criticality. The most limiting postulated accident bounding the cask pit pool has been determined to occur in the spent fuel pool. The postulated accident with respect to the storage configurations assumed in the spent fuel rack criticality analysis is the misplacement of a nominal 4.95 ± 0.05 w/o U-235 fuel assembly into an storage cell location in the Region 2 checkerboard storage arrangement for an irradiated fuel assembly. The amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 due to this fuel misload accident is 700 ppm (Ref. 2).

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The cask pit pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the cask pit pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the cask pit pool.

(continued)

PLANT SYSTEMS **[This page deleted]**

BASES (continued)

APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the cask pit pool.
ACTIONS	Action a: When the concentration of boron in the cask pit pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored along with suspending movement of fuel assemblies. Action a is modified by a provision indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4 is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	Surveillance 4.7.14.1 This Surveillance Requirement verifies that the concentration of boron in the cask pit pool is within the required limit. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no significant replenishment of pool water is expected to take place over such a short period of time. (Ref. 4) Surveillance 4.7.14.2 This Surveillance Requirement verifies that the concentration of boron in the cask pit pool is within the required limit during fuel movement until the final configuration of the assemblies in the storage racks is verified to be correct. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 72 hour Frequency provides additional assurance that the maximum k_{eff} remains below the 0.95 limit under the postulated accident condition. (Ref. 1)

(continued)

PLANT SYSTEMS **[This page deleted]**

BASES (continued)

REFERENCES

1. NRC letter to TVA dated April 28, 1993, "Issuance of Amendments (TAC Nos. M83068 and M83069)"

2. Stanley E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349

3. FSAR, Section 15.4.5

4. K. K. Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NO. 327, 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
REVISED PAGES

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5-5g	5-5g
5-5h	5-5h
5-5i	5-5i
5-5j	5-5j
5-5k	5-5k
5-5l	5-5l
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II. REVISED PAGES

See attached

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

This portion affected by TS Change Request 99-18

This portion affected by TS Change Request 99-18

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

	INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Source Range Nuclear Flux	NOTE 1	0.1 to 1 x 10^5 cps	1
2.	Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3.	Reactor Coolant Temperature - Hot Leg	NOTE 1	0-650°F	1/loop
4.	Pressurizer Pressure	NOTE 1	0-3000 psig	1
5.	Pressurizer Leve1	NOTE 1	0-100%	1
6.	Steam Generator Pressure	NOTE 1	0-1200 psig	1/steam generator
7.	Steam Generator Leve1	NOTE 2 or near Auxilary F. W. Pump	0-100%	1/steam generator
8.	Deleted			
9.	RHR Flow Rate	NOTE 1	0-4500 gpm	1
10.	RHR Temperature	NOTE 1	50-400°F	1
11.	Auxiliary Feedwater Flow Rate	NOTE 1	0-440 gpm	1/steam generator

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7615 and 7960 gallons of borated water,

c. Between 3500 and 3800 ppm of boron,

d. A nitrogen cover-pressure of between 624 and 668 psig, and

e. Power removed from isolation valve when RCS pressure is above 2000 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

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*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between 3600 and 3800 ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

PLANT SYSTEMS

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 This specification has been deleted.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-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. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the Unit 1 Operating Cycle 12.

Sequoyah is authorized to place a maximum of 2256 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 $\pm$ 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel that has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

a. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron.*

b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Arrangements of one or more of three different arrays (Regions) or sub-arrays as illustrated in Figures 5.6-1 and 5.6-1a. These arrangements in the spent fuel storage pool have the following definitions:

   1. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 $\pm$ 0.05 wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 Type A spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 5.6-2 and defined by the equations in Table 5.6-1. The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-2 or Table 5.6-1.

      Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or integral fuel burnable absorber (IFBA). For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 5.6-2.

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*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

5.6 FUEL STORAGE

Restrictions in Region 1

Any of the three sub-arrays illustrated in Figure 5.6-1a may be used in any combination provided that:

A) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-1 and 5.6-2, as appropriate.

B) The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

C) If Region 1 arrays are used in conjunction with Region 3 or Region 4 arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 5.6-1).

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 1, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

2. Region 2 is designed to accommodate Type A or Type T fuel of 4.95±0.05 wt% U-235 initial enrichment burned to at least 30.27 (Type A) or 33.1095 (Type T) 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 in MWD/KgU and cooling time is given by the equations in Table 5.6-3 (Type A) or 5.6-4 (Type T). The minimum required burnups are illustrated in Figure 5.6-3 (Type A) or 5.6-4 (Type T) in terms of the initial enrichment and cooling time.

Restrictions in Region 2

The following restrictions apply to the storage of spent fuel in the Region 2 cells:

A) The spent fuel shall conform to the minimum burnup requirements defined by the equations in Table 5.6-3 or 5.6-4, as appropriate. Linear interpolation between cooling times may be made if desired.

B) For the interface with Region 1 or 4 storage cells, fresh fuel in Region 1 or 4 shall not be stored adjacent to spent fuel assemblies in the Region 2 storage cells.

C) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 2, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

3. Region 3 is designed to accommodate fuel of 4.95±0.05 wt% U-235 initial enrichment (or fuel assemblies of any lower reactivity) in a 2-out-of-4 checkerboard arrangement with water-filled cells. The water-filled cells shall not contain any components bearing any fissile material, but may accommodate miscellaneous items or equipment.

5.6 FUEL STORAGE

Restrictions in Region 3

The following restrictions apply to the storage of fuel in the Region 3 cells:

A) For the interface between Region 3 and Region 1 or 4 storage regions, fresh fuel assemblies shall not be stored adjacent to each other.

B) If miscellaneous non-fissile bearing items or equipment are stored in the water cells of Region 3, the total volume of the miscellaneous items shall be no more than 75% of the storage cell volume.

C) No loose fuel rods or items containing fissile material shall be stored in the water cells of Region 3.

4. Region 4 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 $\pm$ 0.05 wt% U-235 (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with three Type T spent fuel assemblies having burnup and cooling times illustrated in Figure 5.6-5 and defined by the equations in Table 5.6-5. The presence of either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-5 or Table 5.6-5.

   One alternative storage array (or sub-array) is acceptable in Region 4 if the fresh fuel contains rods with gadolinia fuel burnable absorber. For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array is defined by the equations in Table 5.6-6 and illustrated in Figure 5.6-6. For fresh assemblies containing more than eight (8) gadolinia bearing fuel rods, the limiting burnup for eight (8) gadolinia rods shall apply.

Restrictions in Region 4

Any of the two sub-arrays illustrated in Figure 5.6-1a applying to Region 4 storage may be used in any combination provided that:

A) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-5 and 5.6-6, as appropriate.

B) The arrangement of Region 4 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

C) If Region 4 arrays are used in conjunction with Region 1 or 3 arrangements, the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see Figure 5.6-1)

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 4, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

d. An empty cell (or a cell containing non-fissile bearing miscellaneous items displacing no more than 75% of the storage cell volume) is less reactive than any cell containing fuel and therefore may be used as a Region 1, 2, 3, or 4 cell in any arrangement.

e. A nominal concentration of 2000 ppm boron is in the pool water. This concentration of soluble boron provides a margin sufficient to allow timely detection of a boron dilution accident and corrective action before the minimum concentration (700 ppm) required to protect against the most severe postulated fuel handling accident or before the minimum concentration (300 ppm) required to maintain the storage configuration design basis (keff less than 0.95) is reached.

5.6 FUEL STORAGE

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed for fuel enriched to 5.0 weight percent U-235 and shall be maintained with the arrangement of 146 storage locations shown in Figure 5.6-7. The cells shown as empty cells in Figure 5.6-7 shall have physical barriers installed to ensure that inadvertant loading of fuel assemblies into these locations does not occur. This configuration ensures k_{eff} will remain less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 under optimum moderation conditions.

DRAINAGE

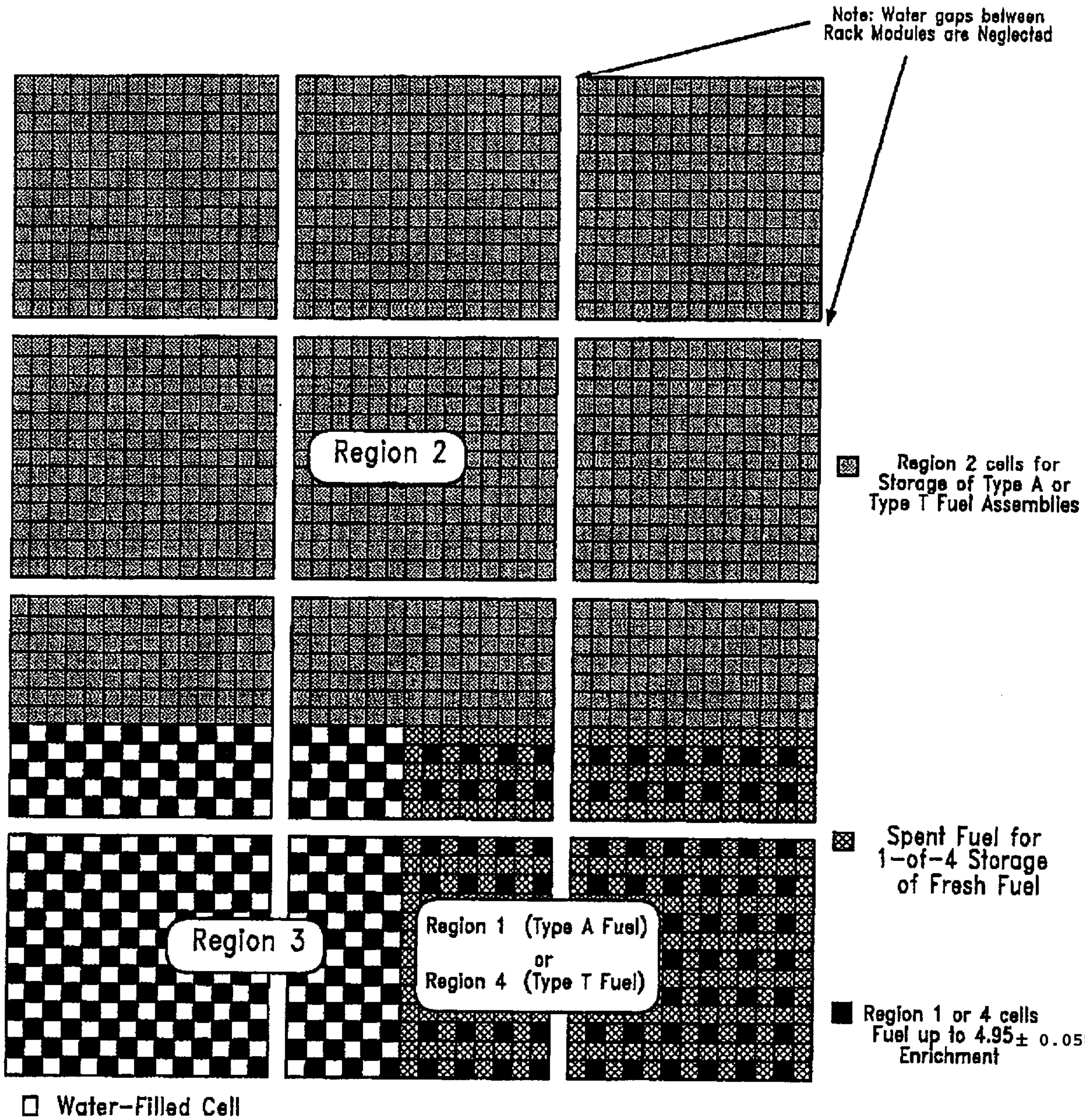
5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies.

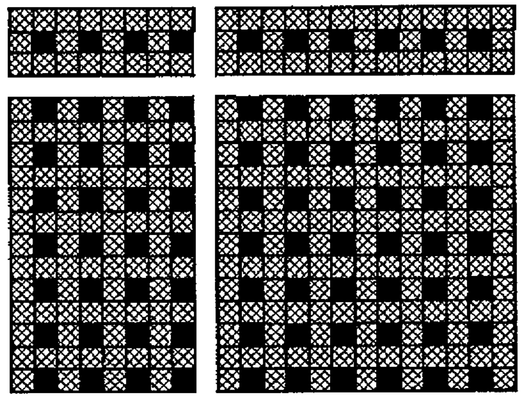
5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.



Note: The edges of the sketch above are not necessarily the edges of the pool. The Regions may appear anywhere in the pool and in any orientation, subject to the restrictions in Design Features 5.6.1.1.c.

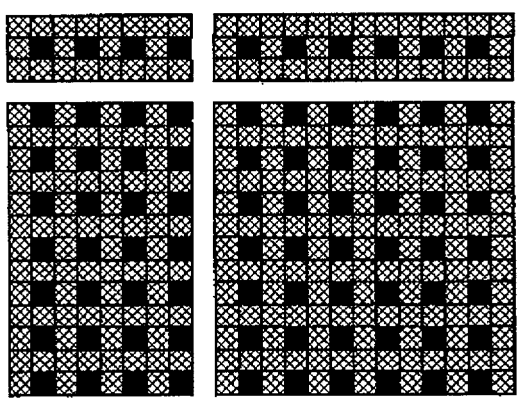
FIG 5.6-1 Arrangements of Fuel Storage Regions in the Sequoyah Spent Fuel Storage Pool



Spent Fuel for Storage in Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

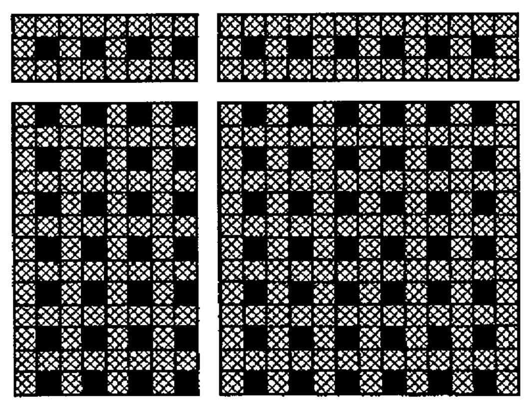
Fresh Fuel for Region 1 or Region 4 cells (No Gd or IFBA)

NOTE: WHEN CREDIT IS TAKEN FOR GADOLINIA RODS IN FRESH ASSEMBLIES THE SPENT FUEL ASSEMBLIES NEED NOT HAVE CONTAINED GADOLINIA RODS..



Spent Fuel for Storage of Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

Fresh Fuel with Gadolinia for Region 1 or Region 4 cells



Spent Fuel for Storage in Region 1 in 1-of-4 Pattern

Fresh Fuel with IFBA Rods for Storage in Region 1 only (1-of-4 Pattern)

Fig. 5.6-1a Acceptable Storage Patterns for Checkerboard Storage of Fresh and Spent Fuel Assemblies in Region 1 or Region 4 – Example

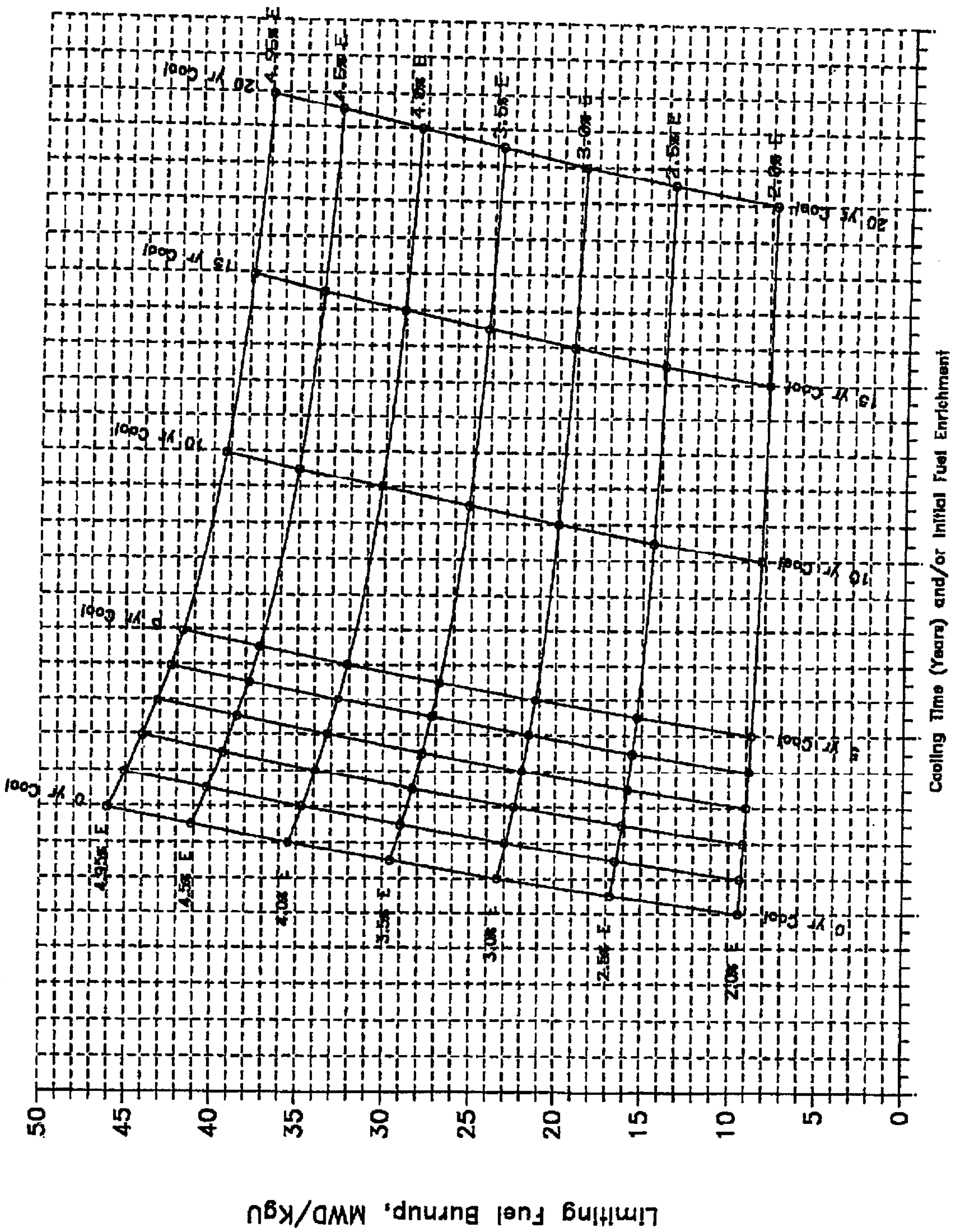


Fig. 5.6-2 3-Dimensional Plot of Minimum Fuel Burnups for Type A Fuel In Region 1 for Enrichments and/or Cooling Times

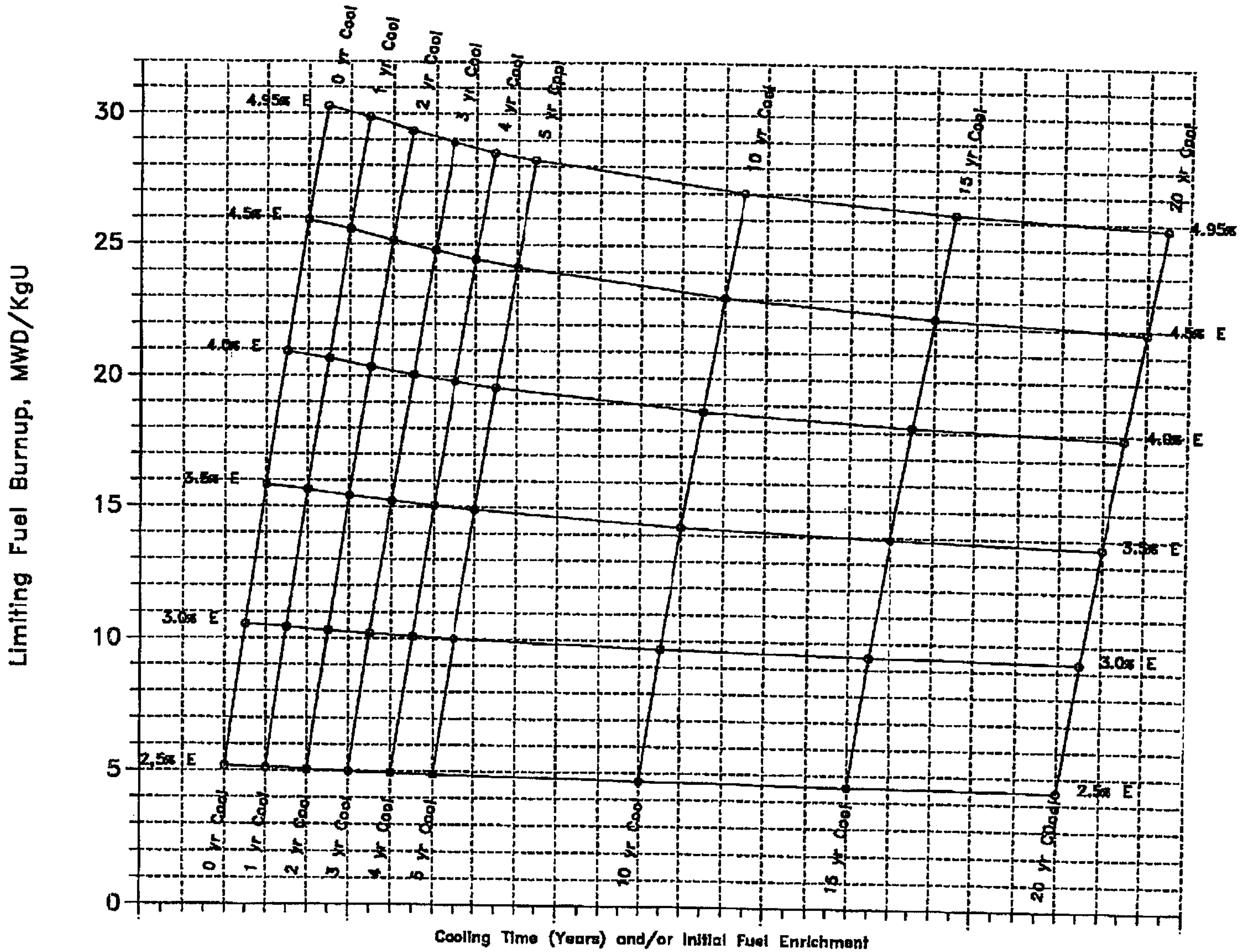


Fig. 5.6-3 3-dimensional Plot of Minimum Fuel Burnups For Type A Fuel in Region 2 for Enrichments and Cooling Times

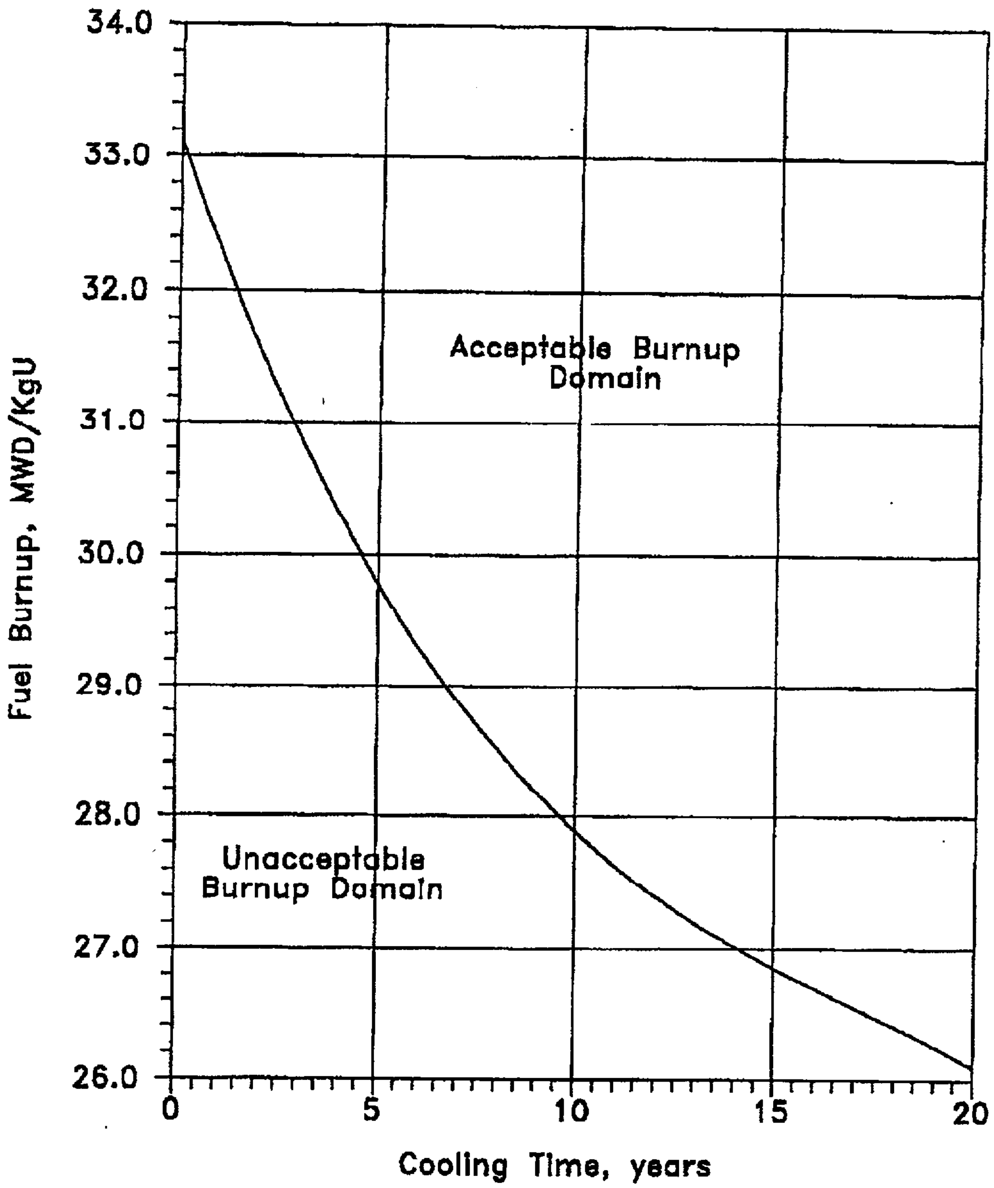


Fig. 5.6–4 Limiting Burnup Requirements in Region 2 for Face Adjacent Storage of Type T Spent Fuel Assemblies

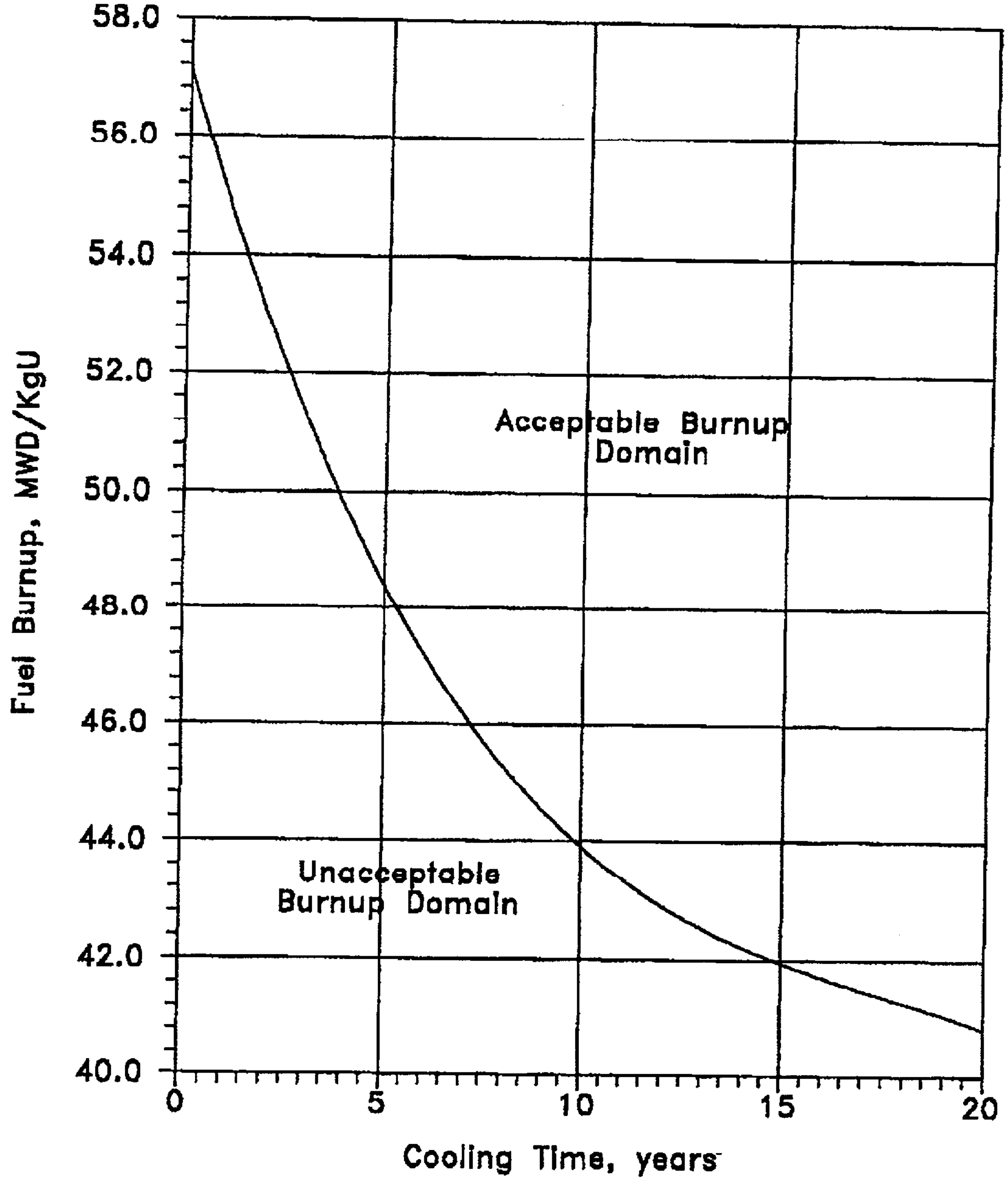


Fig. 5.6-5 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh and 3 Type T Spent Fuel Assemblies

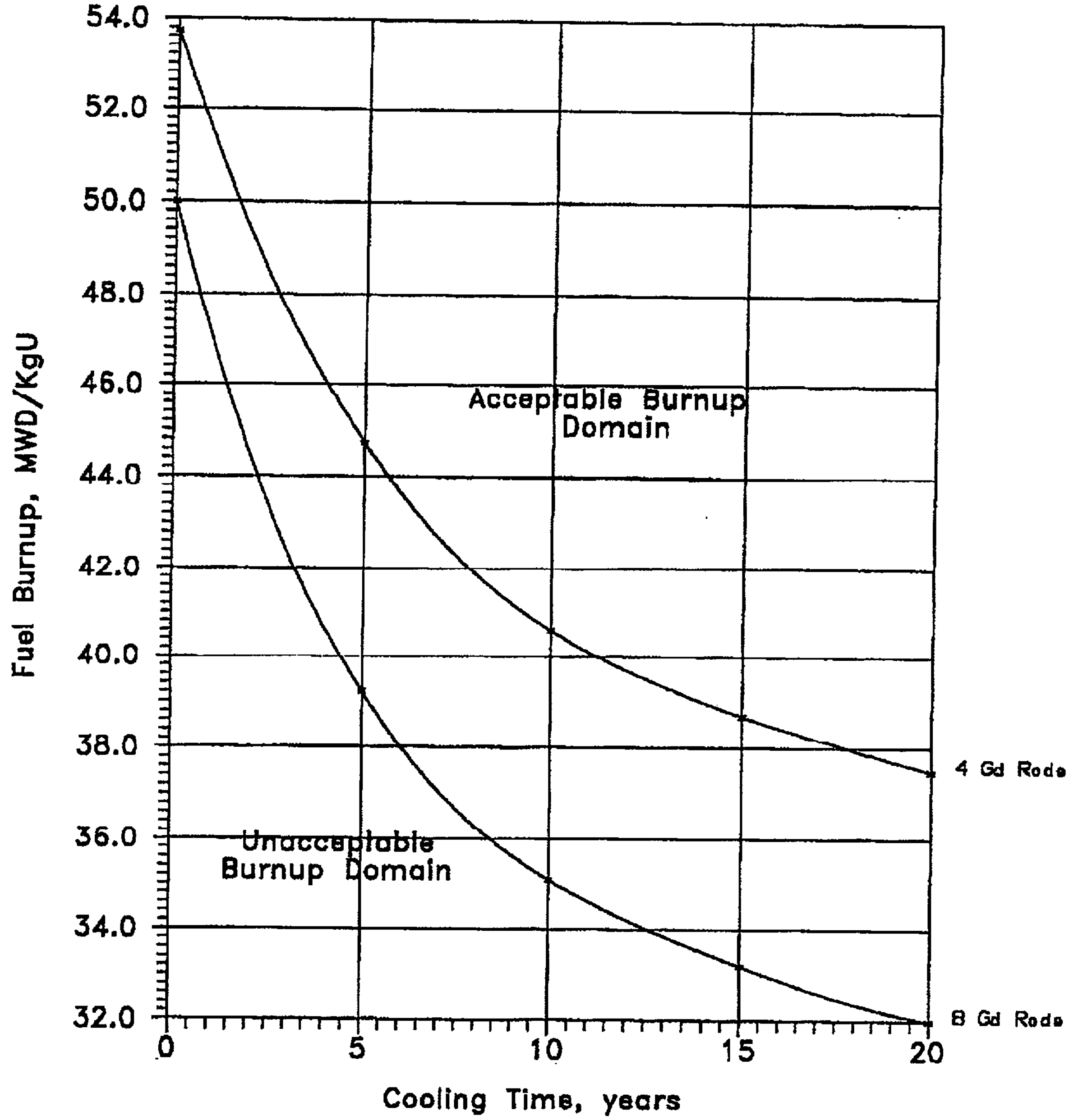
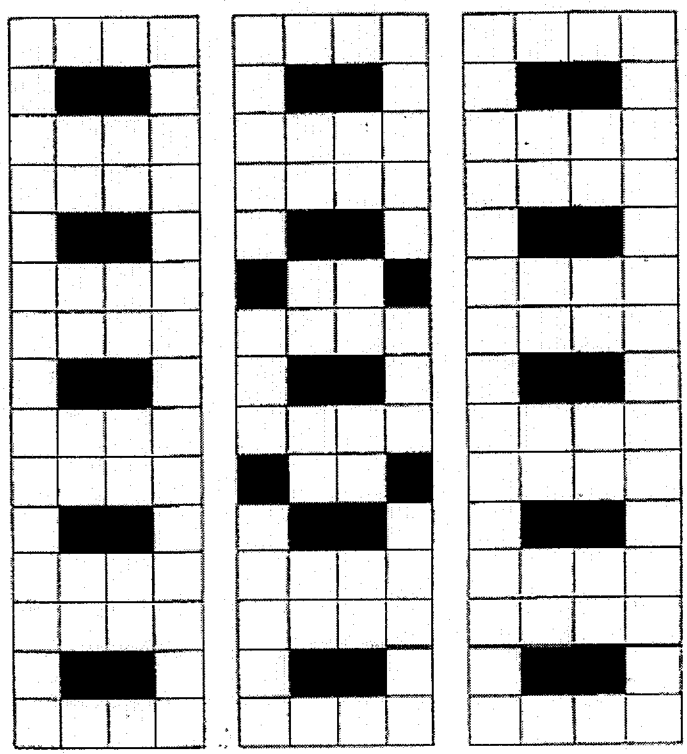


Fig. 5.6-6 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh (with Gadolinia) and 3 Type T Spent Fuel Assemblies



Basic Cell 21 inch X 21 inch

Empty Cell

9 - 4 X 5 Cell Racks

146 / 180 Loading Pattern

Figure 5.6-7
New Fuel Pit Storage Rack Loading Pattern

Table 5.6-1
Region 1 Storage Burnup Restrictions: Checkerboard of 1
Fresh Fuel Assembly (Without Gadolinium or IFBA Rods) and 3 Type A Spent Fuel Assemblies

For Zero Year Cooling Time
Bu (limit) = $-28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
For One Year Cooling Time
Bu (limit) = $-27.3317 + 22.5087 \times E - 2.40586 \times E^2 + 0.164207 \times E^3$
For Two Years Cooling Time
Bu (limit) = $-26.4693 + 21.8404 \times E - 2.31873 \times E^2 + 0.158218 \times E^3$
For Three Years Cooling Time
Bu (limit) = $-25.7404 + 21.2659 \times E - 2.24287 \times E^2 + 0.153018 \times E^3$
For Four Years Cooling Time
Bu (limit) = $-25.1367 + 20.7910 \times E - 2.18484 \times E^2 + 0.1499363 \times E^3$
For Five Years Cooling Time
Bu (limit) = $-24.5981 + 20.3568 \times E - 2.12719 \times E^2 + 0.145431 \times E^3$
For Ten Years Cooling Time
Bu (limit) = $-23.2050 + 19.2969 \times E - 2.06993 \times E^2 + 0.145875 \times E^3$
For Fifteen Years Cooling Time
Bu (limit) = $-22.6098 + 18.8544 \times E - 2.08617 \times E^2 + 0.150473 \times E^3$
For Twenty Years Cooling Time
Bu (limit) = $-22.3017 + 18.622 \times E - 2.11206 \times E^2 + 0.15467 \times E^3$

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-2
Region 1 Storage Burnup Restrictions with Gadolinium or IFBA in Fresh Fuel

With Gadolinium Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 Type A Spent Fuel Assemblies

Zero Year Cooling Time, 0 Gadolinia Rods
Bu (limit) = $- 28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 4 Gadolinia Rods
Bu (limit) = $- 28.4012 + 22.0062 \times E - 2.19268 \times E^2 + 0.143601 \times E^3$
Zero Year Cooling Time, 8 Gadolinia Rods
Bu (limit) = $- 31.4262 + 22.0768 \times E - 2.38845 \times E^2 + 0.164888 \times E^3$

Note: If more that 8 Gadolinium rods per assembly, use the 8 rod correlation

With IFBA Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 Type A Spent Fuel Assemblies

Zero Year Cooling Time, 0 IFBA Rods
Bu (limit) = $- 28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 16 IFBA Rods
Bu (limit) = $- 28.5048 + 21.6411 \times E - 2.15262 \times E^2 + 0.140904 \times E^3$
Zero Year Cooling Time, 32 IFBA Rods
Bu (limit) = $- 31.0949 + 22.0435 \times E - 2.36088 \times E^2 + 0.162229 \times E^3$
Zero Year Cooling Time, 48 IFBA Rods
Bu (limit) = $- 33.1342 + 22.3999 \times E - 2.55367 \times E^2 + 0.18082 \times E^3$
Zero Year Cooling Time, 64 IFBA Rods
Bu (limit) = $- 36.0468 + 24.1492 \times E - 3.11807 \times E^2 + 0.233987 \times E^3$

Note: If more that 64 IFBA rods per assembly, use the correlation for 64 IFBA rods

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-3
Region 2 Storage Burnup Restrictions
For Type A Fuel

Zero Cooling Time Bu (limit) = $- 23.8702 + 12.3026 \times E - 0.275672 \times E^2$
1 Year Cooling Time Bu (limit) = $- 23.6854 + 12.2384 \times E - 0.287498 \times E^2$
2 Years Cooling Time Bu (limit) = $- 23.499 + 12.1873 \times E - 0.305988 \times E^2$
3 Years Cooling Time Bu (limit) = $- 23.3124 + 12.1249 \times E - 0.319566 \times E^2$
4 Years Cooling Time Bu (limit) = $- 23.1589 + 12.0748 \times E - 0.332212 \times E^2$
5 Years Cooling Time Bu (limit) = $- 22.6375 + 11.7906 \times E - 0.307623 \times E^2$
10 Years Cooling Time Bu (limit) = $- 21.7256 + 11.3660 \times E – 0.31029 \times E^2$
15 Years Cooling Time Bu (limit) = $- 21.1160 + 11.0663 \times E - 0.306231 \times E^2$
20 Years Cooling Time Bu (limit) = $- 20.6055 + 10.7906 \times E - 0.29291 \times E^2$

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-4

Face Adjacent Storage of Type T Spent Fuel (Region 2)

$$Bu\ (limit) = 33.1095 - 0.845146 \times CT + 0.0399888 \times CT^2 - 0.000762846 \times CT^3$$

Table 5.6-5

Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel (Region 4: 1 Fresh Assembly and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

$$Bu\ (limit) = 57.118 - 2.13277 \times CT + 0.0772537 \times CT^2 + 0.00127446 \times CT^3 - 9.15855\ E\text{-}5 \times CT^4$$

Table 5.6-6

Gadolinia Credit: Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel (Region 4: 1 Fresh Assembly With Gadolinia and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

4 Gadolinia Rods

$$Bu\ (limit) = 53.73 - 2.5265 \times CT + 0.172283 \times CT^2 - 0.00585995 \times CT^3 + 0.0000766655 \times CT^4$$

8 Gadolinia Rods

$$Bu\ (limit) = 50.00 - 3.26817 \times CT + 0.276117 \times CT^2 - 0.0117934 \times CT^3 + 0.000195334 \times CT^4$$

Note: 1. If more than 8 gadolinia rods per assembly, use the 8 rod correlation
2. BU = Fuel Burnup, MWD/Kg-U; CT = Cooling Time of Spent Fuel Assemblies, Years

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the hydrogen mitigation system, consisting of 68 hydrogen ignitions per unit, is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, 3) corrosion of metals within containment, and 4) tritium and hydrogen that exist in the Tritium Producing Burnable Absorber Rods prior to the accident. These hydrogen control systems are designed to mitigate the effects of an accident as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", Revision 2 dated November 1978. The hydrogen monitors of Specification 3.6.4.1 are part of the accident monitoring instrumentation in Specification 3.3.3.7 and are designated as Type A, Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The operability of at least 66 of 68 ignitors in the hydrogen mitigation system will maintain an effective coverage throughout the containment. This system of ignitors will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1071 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,082,024 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the

B 3.7 PLANT SYSTEMS

B 3/4.7.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION

BASES

BACKGROUND

The spent fuel racks have been analyzed in accordance with the Holtec International methodology contained in Holtec Reports HI - 992349 (Ref. 1) and HI-2012629 (Ref 9). This methodology ensures that the spent fuel rack multiplication factor, k_{eff} is less than or equal to 0.95, as recommended by the NRC guidance contained in NRC Letter to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 and USNRC Internal Memorandum from L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998 (Refs. 2 & 3). The codes, methods, and techniques contained in the methodology are used to satisfy the k_{eff} criterion. The spent fuel storage racks were analyzed using Westinghouse 17x17 V5H fuel assemblies, with enrichments up to 4.95 $\pm$0.05 w/o U-235 and configurations which take credit for checkerboarding, burnup, soluble boron, integral fuel burnable absorbers (such as IFBA or gadolinia), and cooling time to ensure that k_{eff} is maintained $\leq$ 0.95, including uncertainties, tolerances, and accident conditions. The analysis also accounts for the reactivity effects of operating the fuel with discrete burnable poisons (such as burnable poison rod absorbers or tritium producing burnable absorber rods). In addition, the SFP k_{eff} is maintained < 1.0, including uncertainties, tolerances on a 95/95 basis without any soluble boron. Calculations were performed to evaluate the reactivity of fuel types used at SQN. The results show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity, thereby bounding all fuel types utilized and stored at SQN.

In the high density Spent Fuel Rack design (Ref. 9), the spent fuel storage pool is divided into four separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel which has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

Region 1 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 +/- 0.05 wt% U-235, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type A fuel

BASES

BACKGROUND (continued)

assemblies with enrichment, burnup, and cooling times in accordance with Design Feature 5.6.1.1.c.1. Region 2 is designed to accommodate Type A or Type T fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment burned to an assembly average burnup of at least 30.27 MWD/kgU for Type A fuel or 33.1095 MWD/kgU for Type T fuel, or other enrichment with a burnup yielding an equivalent reactivity in the fuel racks in accordance with Design Feature 5.6.1.1.c.2. Region 3 is designed to accommodate fresh fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or fuel assemblies of any lower reactivity in a 2-of-4 checkerboard arrangement with water-filled cells in accordance with Design Feature 5.6.1.1.c.3. Region 4 is designed to accommodate fresh fuel up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type T fuel assemblies with burnup and cooling times in accordance with Design Feature 5.6.1.1.c.4.

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 < 1.0 be evaluated in the absence of soluble boron. Hence, the design of all 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 (Ref. 5) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks may therefore be achieved by controlling the location of each assembly in accordance with Design Features 5.6 FUEL STORAGE. During fuel movement, it is necessary to perform Surveillance Requirement 4.7.13.2.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of any one of the three regions. Examples of these accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in all regions. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack

BASES

APPLICABLE SAFETY ANALYSES (continued)

criticality analysis is the misplacement of a nominal 4.95 ± 0.05 w/o U-235 fresh fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement. The amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 due to this fuel misload accident is 700 ppm (Ref. 1 and Ref. 9).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the k_{eff} design basis limit of 0.95 (Ref. 6). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event.

The concentration of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The spent fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the spent fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 7. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

Action a:

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored along with suspending movement of fuel assemblies.

Action a is modified by a provision indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4 is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

4.7.13.1

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no significant replenishment of pool water is expected to take place over such a short period of time. (Ref. 6)

4.7.13.2

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit during fuel movement until the final configuration of the assemblies in the storage racks is verified to be correct. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 72 hour Frequency provides additional assurance that the maximum k_{eff} remains below the 0.95 limit under the postulated accident condition. (Ref. 1, 8, and 9)

REFERENCES

1. Stanley E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349
2. B.K. Grimes (NRC GL78011), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978
3. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998
4. UFSAR, Section 4.3.2.7, "Critically of Fuel Assemblies"
5. 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).
6. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302
7. FSAR, Section 15.4.5
8. NRC letter to TVA dated August 1, 1990, " Increase Fuel Enrichment to 5.0 Weight Percent (TAC Nos. 76074, 76075, 76774, 76775) (TS 90-12) - Sequoyah Nuclear Plant, Units 1 and 2"
9. Stanley E. Turner (Holtec International, "Evaluation of the Effect of the Use of Tritium Producing Burnable Absorber Rods (TPBARS) on Fuel Storage Requirements," HI-2012629

PLANT SYSTEMS

BASES

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

This specification is deleted.

Pages B3/4 7-13 through B3/4 7-15 are deleted.

(continued)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

This portion affected by TS Change Request 99-18

BASES

SECTION | PAGE

INDEX

DESIGN FEATURES

SECTION PAGE

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Source Range Nuclear Flux	NOTE 1	0.1 to 1 x 10^5 cps	1
2. Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3. Reactor Coolant Temperature - Hot Leg	NOTE 1	0-650°F	1/loop
4. Pressurizer Pressure	NOTE 1	0-3000 psig	1
5. Pressurizer Level	NOTE 1	0-100%	1
6. Steam Generator Pressure	NOTE 1	0-1200 psig	1/steam generator
7. Steam Generator Level	NOTE 2 or near Auxilary F. W. Pump	0-100%	1/steam generator
8. Deleted			
9. RHR Flow Rate	NOTE 1	0-4500 gpm	1
10. RHR Temperature	NOTE 1	50-400°F	1
11. Auxiliary Feedwater Flow Rate	NOTE 1	0-440 gpm	1/steam generator

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7615 and 7960 gallons of borated water,

c. Between 3500 and 3800 ppm of boron,

d. A nitrogen cover-pressure of between 624 and 668 psig, and

e. Power removed from isolation valve when RCS pressure is above 2000 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to 1000 psig or less within the following 6 hours.

* Pressurizer pressure above 1000 psig.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between 3600 and 3800 ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

PLANT SYSTEMS

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 This specification has been deleted.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-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. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor, as described in the Framatome Cogema Fuels Report BAW-2328, beginning with the Unit 2 Operating Cycle 10 core.

Sequoyah is authorized to place a maximum of 2256 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operations.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel that has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

a. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron.*

b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Arrangements of one or more of three different arrays (Regions) or sub-arrays as illustrated in Figures 5.6-1 and 5.6-1a. These arrangements in the spent fuel storage pool have the following definitions:

    1. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with 3 Type A spent fuel assemblies with enrichment-burnup and cooling times illustrated in Figure 5.6-2 and defined by the equations in Table 5.6-1. The presence of a removable, non-fissile insert such as a burnable poison rod assembly (BPRA) or either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-2 or Table 5.6-1.

       Two alternative storage arrays (or sub-arrays) are acceptable in Region 1 if the fresh fuel assemblies contain rods with either gadolinia or integral fuel burnable absorber (IFBA). For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array are defined by the equations in Table 5.6-2.

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*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

5.6 FUEL STORAGE

Restrictions in Region 1

Any of the three sub-arrays illustrated in Figure 5.6-1a may be used in any combination provided that:

A) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-1 and 5.6-2, as appropriate.

B) The arrangement of Region 1 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

C) If Region 1 arrays are used in conjunction with Region 3 or Region 4 arrangements (see below), the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see also Figure 5.6-1).

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 1, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

2. Region 2 is designed to accommodate Type A or Type T fuel of 4.95±0.05 wt% U-235 initial enrichment burned to at least 30.27 (Type A) or 33.1095 (Type T) 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 in MWD/KgU and cooling time is given by the equations in Table 5.6-3 (Type A) or 5.6-4 (Type T). The minimum required burnups are illustrated in Figure 5.6-3 (Type A) or 5.6-4 (Type T) in terms of the initial enrichment and cooling time.

Restrictions in Region 2

The following restrictions apply to the storage of spent fuel in the Region 2 cells:

A) The spent fuel shall conform to the minimum burnup requirements defined by the equations in Table 5.6-3 or 5.6-4, as appropriate. Linear interpolation between cooling times may be made if desired.

B) For the interface with Region 1 or 4 storage cells, fresh fuel in Region 1 or 4 shall not be stored adjacent to spent fuel assemblies in the Region 2 storage cells.

C) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 2, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.

3. Region 3 is designed to accommodate fuel of 4.95±0.05 wt% U-235 initial enrichment (or fuel assemblies of any lower reactivity) in a 2-out-of-4 checkerboard arrangement with water-filled cells. The water-filled cells shall not contain any components bearing any fissile material, but may accommodate miscellaneous items or equipment.

5.6 FUEL STORAGE

Restrictions in Region 3

The following restrictions apply to the storage of fuel in the Region 3 cells:

A) For the interface between Region 3 and Region 1 or 4 storage regions, fresh fuel assemblies shall not be stored adjacent to each other.

B) If miscellaneous non-fissile bearing items or equipment are stored in the water cells of Region 3, the total volume of the miscellaneous items shall be no more than 75% of the storage cell volume.

C) No loose fuel rods or items containing fissile material shall be stored in the water cells of Region 3.

4. Region 4 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 $\pm$ 0.05 wt% U-235 (or spent fuel regardless of the fuel burnup), in a 1-in-4 checkerboard arrangement of 1 fresh assembly with three Type T spent fuel assemblies having burnup and cooling times illustrated in Figure 5.6-5 and defined by the equations in Table 5.6-5. The presence of either gadolinia or integral fuel burnable absorber (IFBA) in a fresh fuel assembly does not affect the applicability of Figure 5.6-5 or Table 5.6-5.

One alternative storage array (or sub-array) is acceptable in Region 4 if the fresh fuel contains rods with gadolinia fuel burnable absorber. For these types of assemblies, the minimum burnup of the spent fuel in the 1-of-4 sub-array is defined by the equations in Table 5.6-6 and illustrated in Figure 5.6-6. For fresh assemblies containing more than eight (8) gadolinia bearing fuel rods, the limiting burnup for eight (8) gadolinia rods shall apply.

Restrictions in Region 4

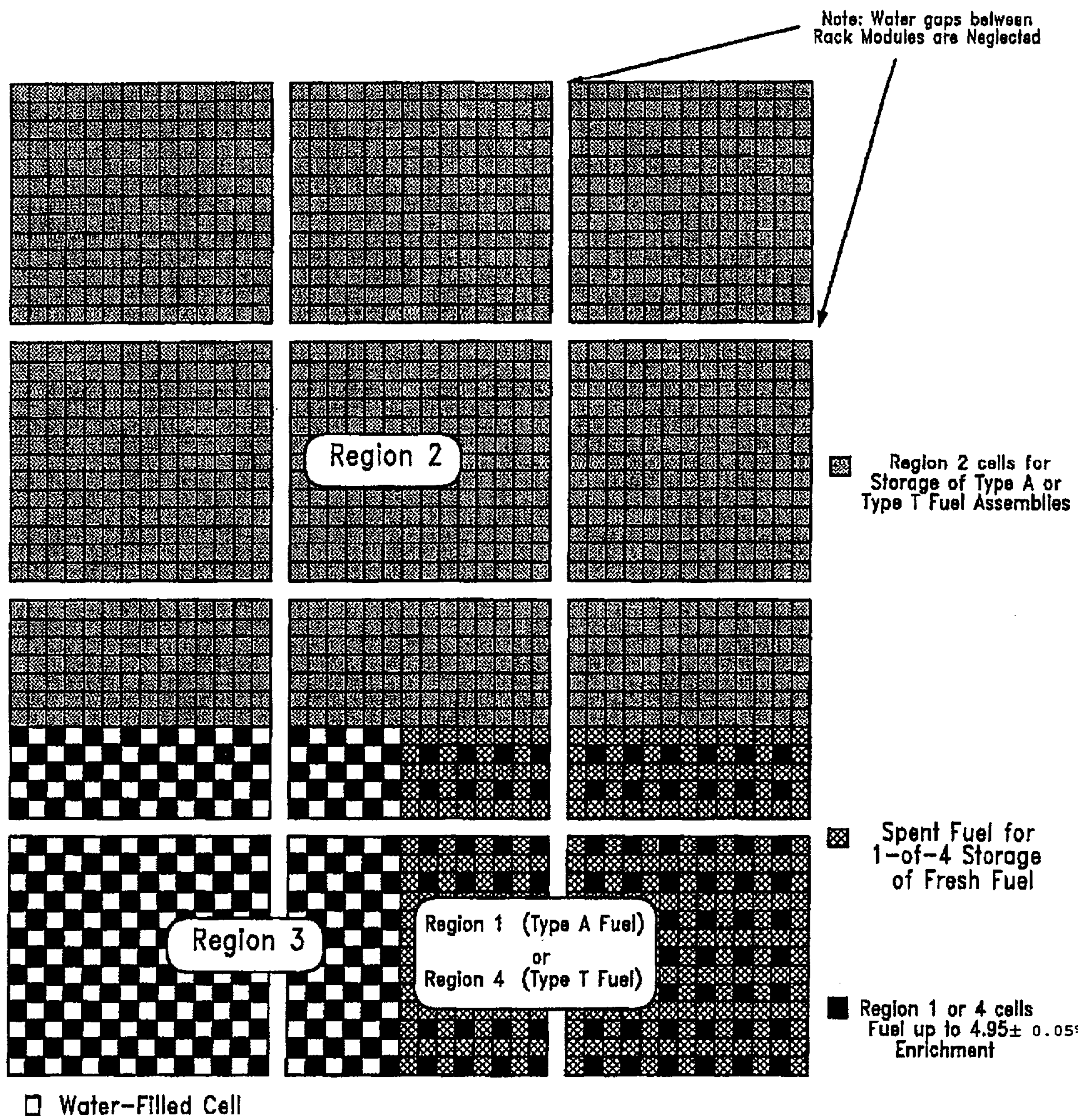
Any of the two sub-arrays illustrated in Figure 5.6-1a applying to Region 4 storage may be used in any combination provided that:

A) Each sub-array of 4 fuel assemblies includes, in addition to the fresh fuel assembly, 3 assemblies with enrichment and minimum burnup requirements defined by the equations in Tables 5.6-5 and 5.6-6, as appropriate.

B) The arrangement of Region 4 sub-arrays must not allow a configuration with fresh assemblies adjacent to each other.

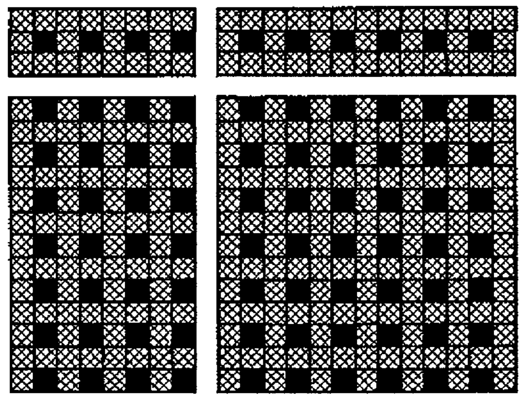
C) If Region 4 arrays are used in conjunction with Region 1 or 3 arrangements, the arrangements shall not allow fresh fuel assemblies to be adjacent to each other (see Figure 5.6-1)

D) If miscellaneous non-fissile bearing items or equipment are stored in cells of Region 4, the total volume of the miscellaneous items shall be no more than 75% of the total storage cell volume.



Note: The edges of the sketch above are not necessarily the edges of the pool. The Regions may appear anywhere in the pool and in any orientation, subject to the restrictions in Design Features 5.6.1.1.c.

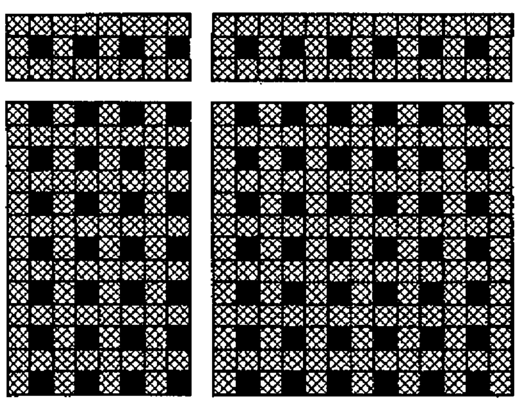
FIG 5.6-1 Arrangements of Fuel Storage Regions in the Sequoyah Spent Fuel Storage Pool



Spent Fuel for Storage in Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

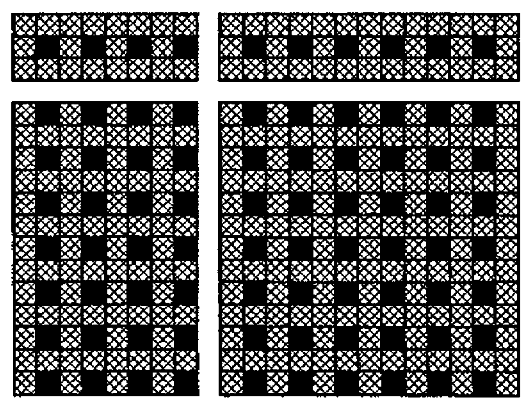
Fresh Fuel for Region 1 or Region 4 cells (No Gd or IFBA)

NOTE: WHEN CREDIT IS TAKEN FOR GADOLINIA RODS IN FRESH ASSEMBLIES THE SPENT FUEL ASSEMBLIES NEED NOT HAVE CONTAINED GADOLINIA RODS..



Spent Fuel for Storage of Region 1 (Type A Fuel) or Region 4 (Type T Fuel) in 1-of-4 Pattern

Fresh Fuel with Gadolinia for Region 1 or Region 4 cells



Spent Fuel for Storage in Region 1 in 1-of-4 Pattern

Fresh Fuel with IFBA Rods for Storage in Region 1 only (1-of-4 Pattern)

Fig. 5.6-1a Acceptable Storage Patterns for Checkerboard Storage of Fresh and Spent Fuel Assemblies in Region 1 or Region 4 – Example

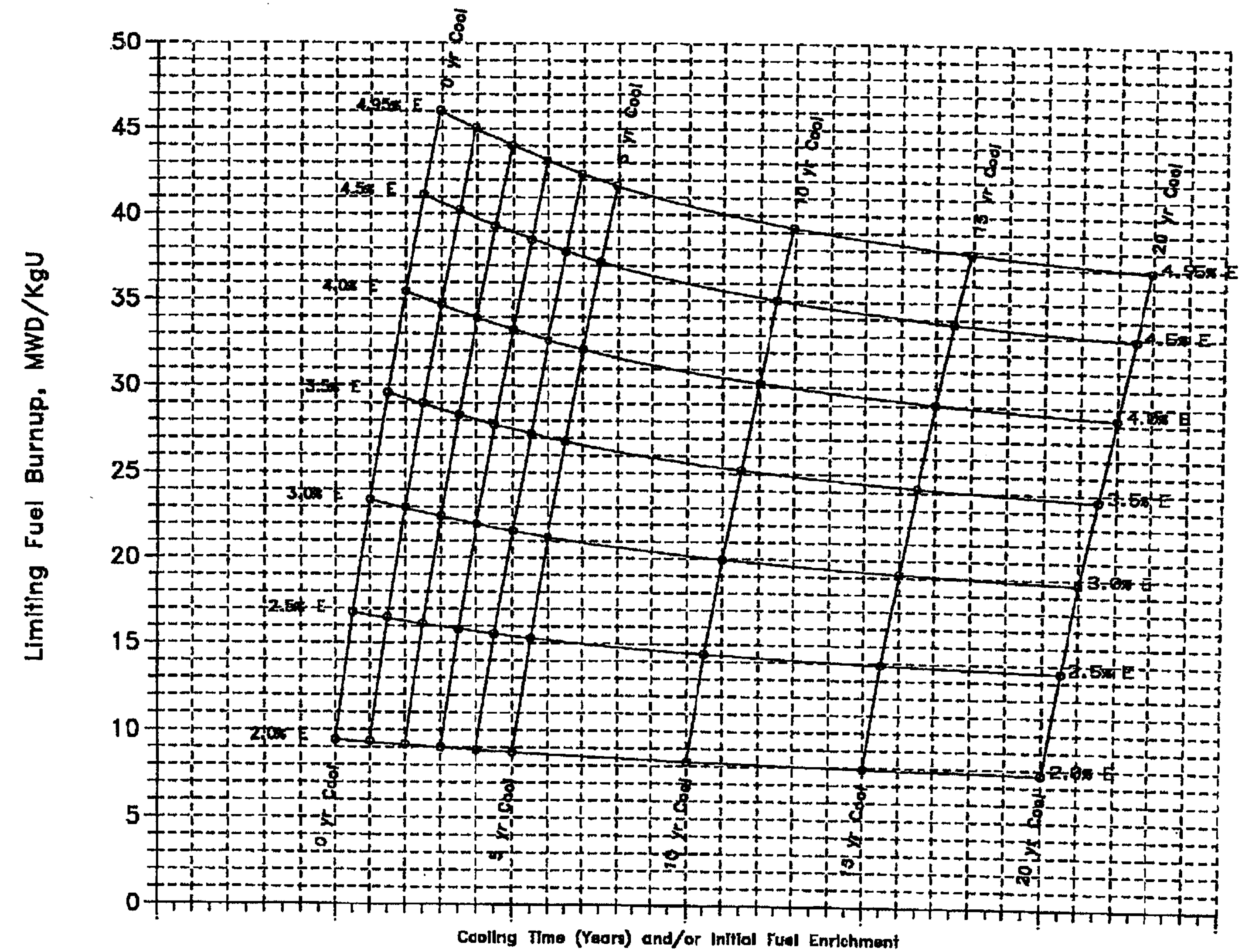


Fig. 5.6-2 3-Dimensional Plot of Minimum Fuel Burnups for Type A Fuel in Region 1 for Enrichments and/or Cooling Times

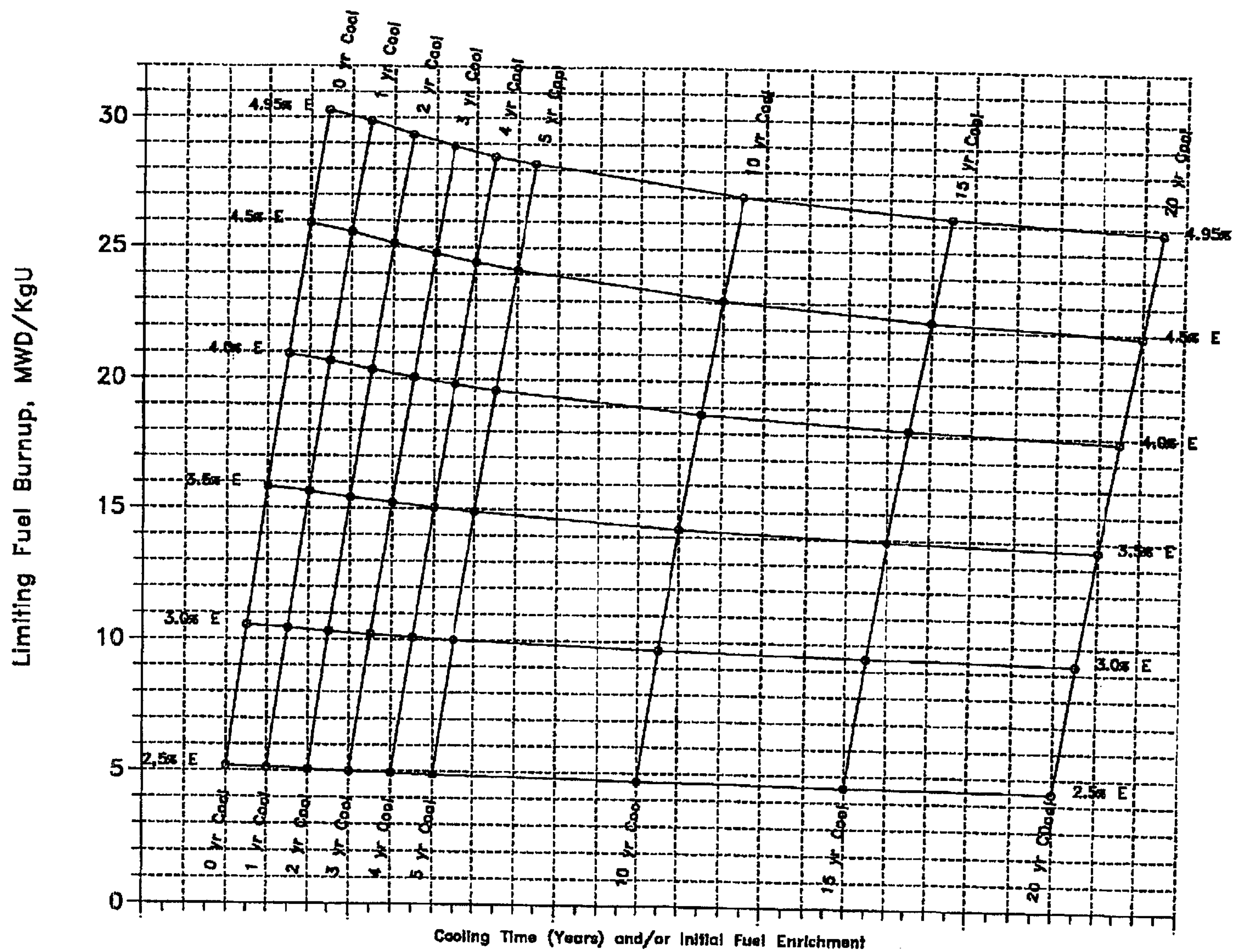


Fig. 5.6-3 3-dimensional Plot of Minimum Fuel Burnups For Type A Fuel in Region 2 for Enrichments and Cooling Times

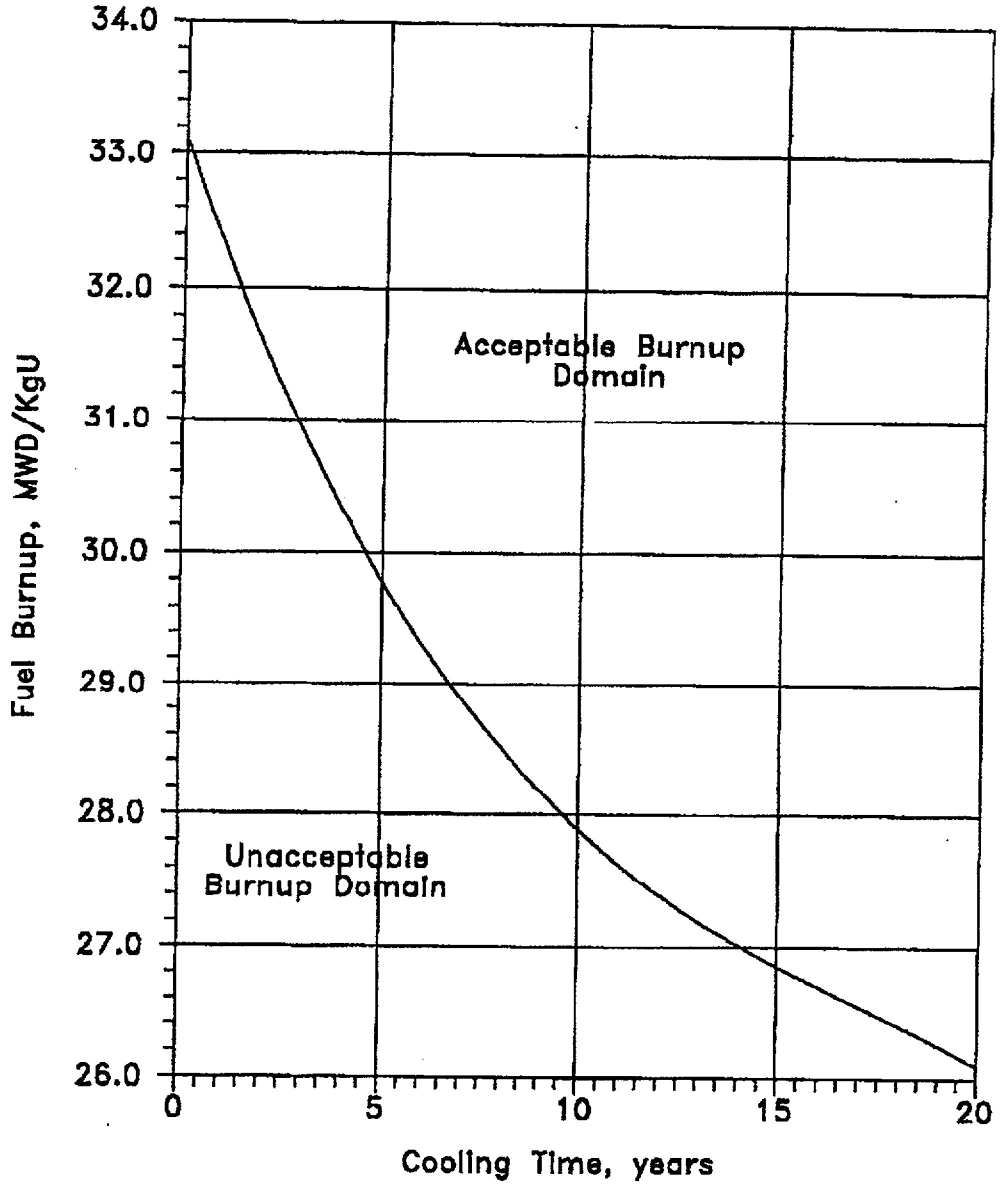


Fig. 5.6–4 Limiting Burnup Requirements in Region 2 for Face Adjacent Storage of Type T Spent Fuel Assemblies

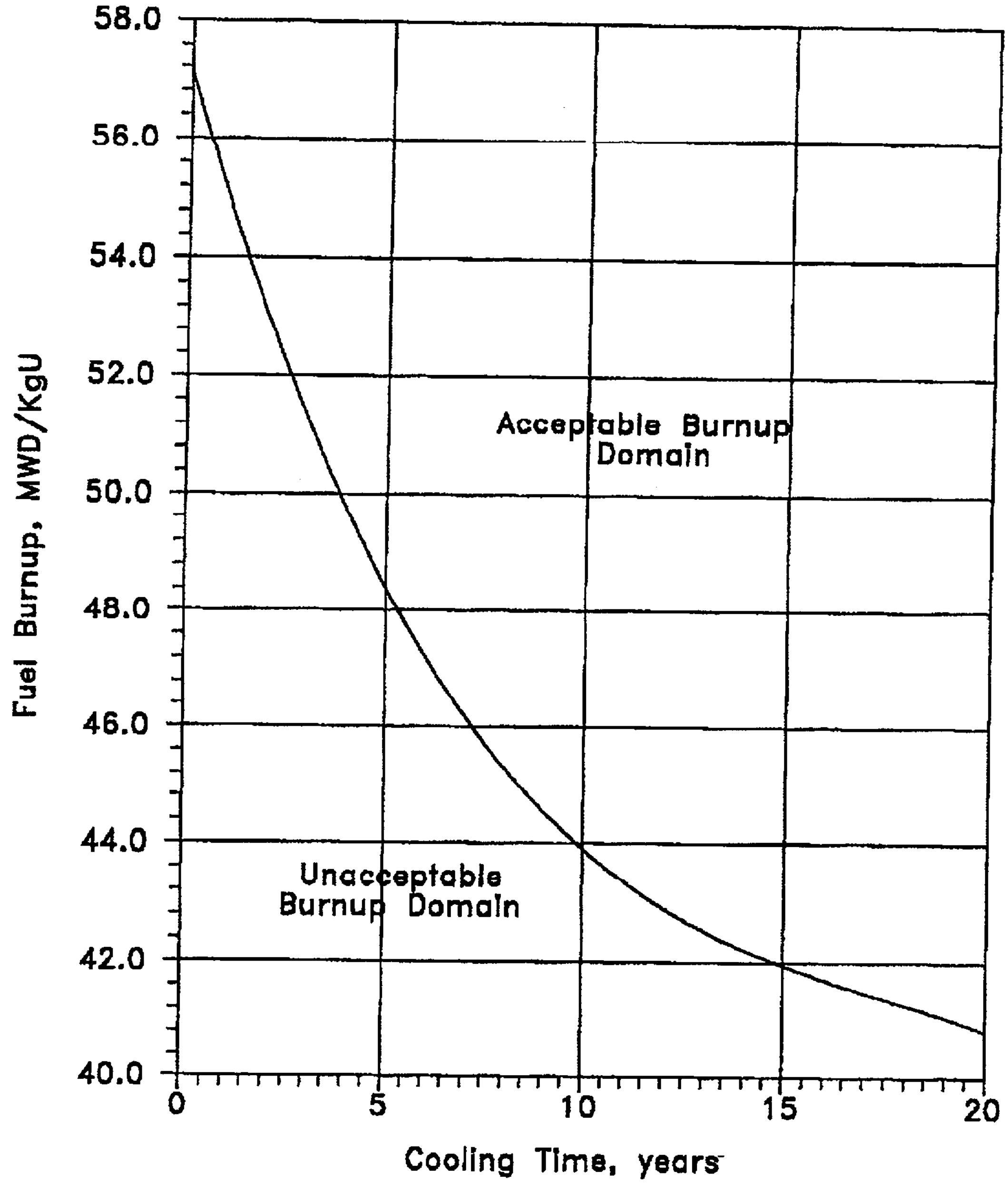


Fig. 5.6–5 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh and 3 Type T Spent Fuel Assemblies

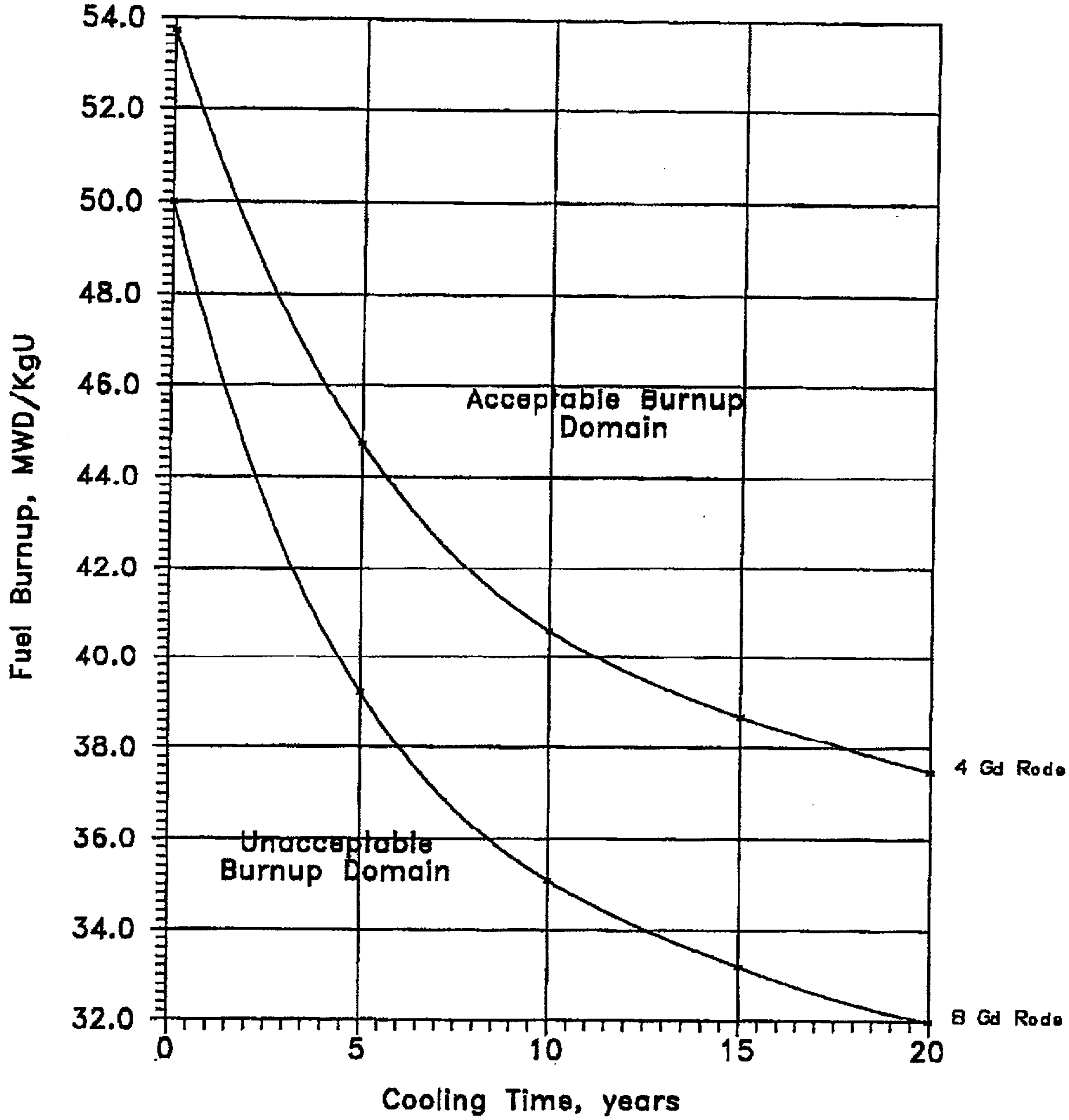


Fig. 5.6–6 Limiting Burnup Requirements in Region 4, Checkerboard Array of 1 Fresh (with Gadolinia) and 3 Type T Spent Fuel Assemblies

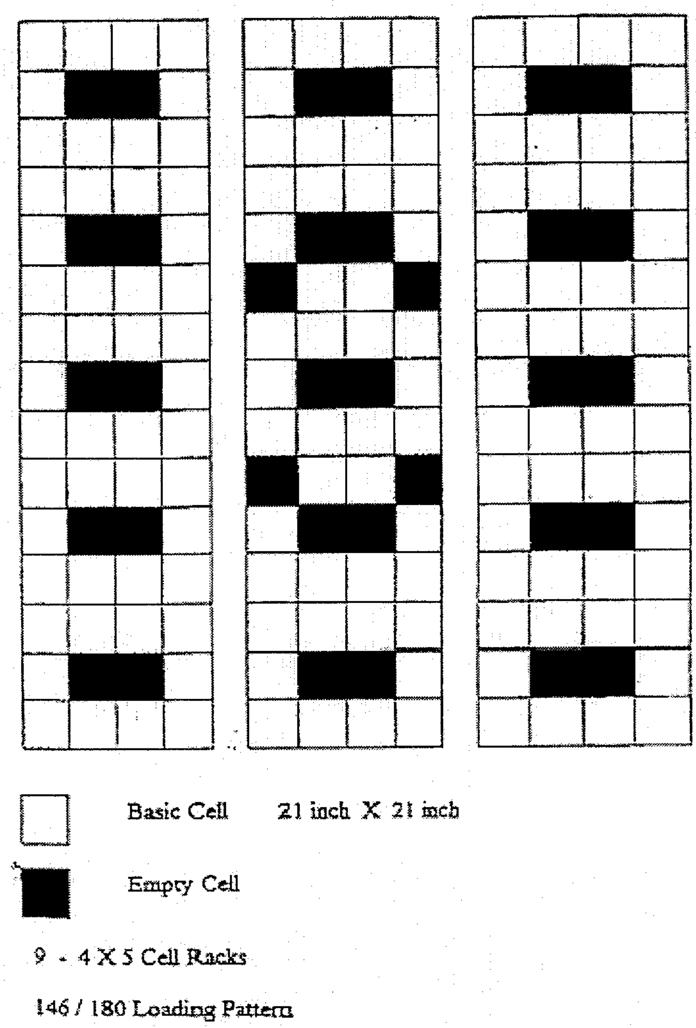


Figure 5.6-7
New Fuel Pit Storage Rack Loading Pattern

Table 5.6-1
Region 1 Storage Burnup Restrictions: Checkerboard of 1
Fresh Fuel Assembly (Without Gadolinium or IFBA Rods) and 3 Type A Spent Fuel Assemblies

For Zero Year Cooling Time
Bu (limit) = $-28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
For One Year Cooling Time
Bu (limit) = $-27.3317 + 22.5087 \times E - 2.40586 \times E^2 + 0.164207 \times E^3$
For Two Years Cooling Time
Bu (limit) = $-26.4693 + 21.8404 \times E - 2.31873 \times E^2 + 0.158218 \times E^3$
For Three Years Cooling Time
Bu (limit) = $-25.7404 + 21.2659 \times E - 2.24287 \times E^2 + 0.153018 \times E^3$
For Four Years Cooling Time
Bu (limit) = $-25.1367 + 20.7910 \times E - 2.18484 \times E^2 + 0.1499363 \times E^3$
For Five Years Cooling Time
Bu (limit) = $-24.5981 + 20.3568 \times E - 2.12719 \times E^2 + 0.145431 \times E^3$
For Ten Years Cooling Time
Bu (limit) = $-23.2050 + 19.2969 \times E - 2.06993 \times E^2 + 0.145875 \times E^3$
For Fifteen Years Cooling Time
Bu (limit) = $-22.6098 + 18.8544 \times E - 2.08617 \times E^2 + 0.150473 \times E^3$
For Twenty Years Cooling Time
Bu (limit) = $-22.3017 + 18.622 \times E - 2.11206 \times E^2 + 0.15467 \times E^3$

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-2
Region 1 Storage Burnup Restrictions with Gadolinium or IFBA in Fresh Fuel

With Gadolinium Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 Type A Spent Fuel Assemblies

Zero Year Cooling Time, 0 Gadolinia Rods Bu (limit) = $- 28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 4 Gadolinia Rods Bu (limit) = $- 28.4012 + 22.0062 \times E - 2.19268 \times E^2 + 0.143601 \times E^3$
Zero Year Cooling Time, 8 Gadolinia Rods Bu (limit) = $- 31.4262 + 22.0768 \times E - 2.38845 \times E^2 + 0.164888 \times E^3$

Note: If more that 8 Gadolinium rods per assembly, use the 8 rod correlation

With IFBA Credit: Checkerboard of 1 Fresh Fuel Assembly with 3 Type A Spent Fuel Assemblies

Zero Year Cooling Time, 0 IFBA Rods Bu (limit) = $- 28.1868 + 23.0765 \times E - 2.46264 \times E^2 + 0.167868 \times E^3$
Zero Year Cooling Time, 16 IFBA Rods Bu (limit) = $- 28.5048 + 21.6411 \times E - 2.15262 \times E^2 + 0.140904 \times E^3$
Zero Year Cooling Time, 32 IFBA Rods Bu (limit) = $- 31.0949 + 22.0435 \times E - 2.36088 \times E^2 + 0.162229 \times E^3$
Zero Year Cooling Time, 48 IFBA Rods Bu (limit) = $- 33.1342 + 22.3999 \times E - 2.55367 \times E^2 + 0.18082 \times E^3$
Zero Year Cooling Time, 64 IFBA Rods Bu (limit) = $- 36.0468 + 24.1492 \times E - 3.11807 \times E^2 + 0.233987 \times E^3$

Note: If more that 64 IFBA rods per assembly, use the correlation for 64 IFBA rods

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-3
Region 2 Storage Burnup Restrictions
For Type A Fuel

Zero Cooling Time $Bu\ (limit) = -23.8702 + 12.3026 \times E - 0.275672 \times E^2$
1 Year Cooling Time $Bu\ (limit) = -23.6854 + 12.2384 \times E - 0.287498 \times E^2$
2 Years Cooling Time $Bu\ (limit) = -23.499 + 12.1873 \times E - 0.305988 \times E^2$
3 Years Cooling Time $Bu\ (limit) = -23.3124 + 12.1249 \times E - 0.319566 \times E^2$
4 Years Cooling Time $Bu\ (limit) = -23.1589 + 12.0748 \times E - 0.332212 \times E^2$
5 Years Cooling Time $Bu\ (limit) = -22.6375 + 11.7906 \times E - 0.307623 \times E^2$
10 Years Cooling Time $Bu\ (limit) = -21.7256 + 11.3660 \times E - 0.31029 \times E^2$
15 Years Cooling Time $Bu\ (limit) = -21.1160 + 11.0663 \times E - 0.306231 \times E^2$
20 Years Cooling Time $Bu\ (limit) = -20.6055 + 10.7906 \times E - 0.29291 \times E^2$

Note: E = initial enrichment in the axial zone of highest enrichment (wt% U-235)

Table 5.6-4
Face Adjacent Storage of Type T Spent Fuel (Region 2)

$$Bu\ (limit) = 33.1095 - 0.845146 \times CT + 0.0399888 \times CT^2 - 0.000762846 \times CT^3$$

Table 5.6-5
Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel
(Region 4: 1 Fresh Assembly and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

$$Bu\ (limit) = 57.118 - 2.13277 \times CT + 0.0772537 \times CT^2 + 0.00127446 \times CT^3 - 9.15855\ E\text{-}5 \times CT^4$$

Table 5.6-6
Gadolinia Credit: Limiting Burnup For Checkerboard Storage of Fresh and Type T Spent Fuel
(Region 4: 1 Fresh Assembly With Gadolinia and 3 Spent Fuel Assemblies in a 2X2 Arrangement)

4 Gadolinia Rods

$$Bu\ (limit) = 53.73 - 2.5265 \times CT + 0.172283 \times CT^2 - 0.00585995 \times CT^3 + 0.0000766655 \times CT^4$$

8 Gadolinia Rods

$$Bu\ (limit) = 50.00 - 3.26817 \times CT + 0.276117 \times CT^2 - 0.0117934 \times CT^3 + 0.000195334 \times CT^4$$

Note: 1. If more than 8 gadolinia rods per assembly, use the 8 rod correlation
2. BU = Fuel Burnup, MWD/Kg-U; CT = Cooling Time of Spent Fuel Assemblies, Years

d. An empty cell (or a cell containing non-fissile bearing miscellaneous items displacing no more than 75% of the storage cell volume) is less reactive than any cell containing fuel and therefore may be used as a Region 1, 2, 3, or 4 cell in any arrangement.

e. A nominal concentration of 2000 ppm boron is in the pool water. This concentration of soluble boron provides a margin sufficient to allow timely detection of a boron dilution accident and corrective action before the minimum concentration (700 ppm) required to protect against the most severe postulated fuel handling accident or before the minimum concentration (300 ppm) required to maintain the storage configuration design basis (keff less than 0.95) is reached.

5.6 FUEL STORAGE

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed for fuel enriched to 5.0 weight percent U-235 and shall be maintained with the arrangement of 146 storage locations shown in Figure 5.6-7. The cells shown as empty cells in Figure 5.6-7 shall have physical barriers installed to ensure that inadvertant loading of fuel assemblies into these locations does not occur. This configuration ensures k_{eff} will remain less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 under optimum moderation conditions.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the hydrogen mitigation system, consisting of 68 hydrogen ignitions per unit, is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, 3) corrosion of metals within containment, and 4) tritium and hydrogen that exist in the Tritium Producing Burnable Absorber Rods prior to the accident. These hydrogen control systems are designed to mitigate the effects of an accident as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", Revision 2 dated November 1978. The hydrogen monitors of Specification 3.6.4.1 are part of the accident monitoring instrumentation in Specification 3.3.3.7 and are designated as Type A, Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The operability of at least 66 of 68 igniters in the hydrogen control distributed ignition system will maintain an effective coverage throughout the containment. This system of ignitors will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1071 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,082,024 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the

B 3.7 PLANT SYSTEMS

B 3/4.7.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION

BASES

BACKGROUND

The spent fuel racks have been analyzed in accordance with the Holtec International methodology contained in Holtec Reports HI - 992349 (Ref. 1) and HI-2012629 (Ref 9). This methodology ensures that the spent fuel rack multiplication factor, k_{eff} is less than or equal to 0.95, as recommended by the NRC guidance contained in NRC Letter to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 and USNRC Internal Memorandum from L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998 (Refs. 2 & 3). The codes, methods, and techniques contained in the methodology are used to satisfy the k_{eff} criterion. The spent fuel storage racks were analyzed using Westinghouse 17x17 V5H fuel assemblies, with enrichments up to 4.95 ±0.05 w/o U-235 and configurations which take credit for checkerboarding, burnup, soluble boron, integral fuel burnable absorbers (such as IFBA or gadolinia), and cooling time to ensure that k_{eff} is maintained ≤ 0.95, including uncertainties, tolerances, and accident conditions. The analysis also accounts for the reactivity effects of operating the fuel with discrete burnable poisons (such as burnable poison rod absorbers or tritium producing burnable absorber rods). In addition, the SFP k_{eff} is maintained < 1.0, including uncertainties, tolerances on a 95/95 basis without any soluble boron. Calculations were performed to evaluate the reactivity of fuel types used at SQN. The results show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity, thereby bounding all fuel types utilized and stored at SQN.

In the high density Spent Fuel Rack design (Ref. 9), the spent fuel storage pool is divided into four separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. For convenience of reference, the following definitions apply:

Type A fuel refers to spent fuel assemblies which have not contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Type T fuel refers to spent fuel assemblies which have contained tritium producing burnable absorber rods (TPBAR's) during in-core operation.

Fresh fuel refers to unirradiated Type A or Type T fuel or irradiated Type A or Type T fuel which has not attained sufficient burnup to meet spent fuel requirements.

Cooling time is defined as the period since reactor shutdown at the end of the last operating cycle for the discharged spent fuel assembly.

Region 1 is designed to accommodate fresh fuel with a maximum enrichment of 4.95 +/- 0.05 wt% U-235, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type A fuel

BACKGROUND (continued)

assemblies with enrichment, burnup, and cooling times in accordance with Design Feature 5.6.1.1.c.1. Region 2 is designed to accommodate Type A or Type T fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment burned to an assembly average burnup of at least 30.27 MWD/kgU for Type A fuel or 33.1095 MWD/kgU for Type T fuel, or other enrichment with a burnup yielding an equivalent reactivity in the fuel racks in accordance with Design Feature 5.6.1.1.c.2. Region 3 is designed to accommodate fresh fuel of up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or fuel assemblies of any lower reactivity in a 2-of-4 checkerboard arrangement with water-filled cells in accordance with Design Feature 5.6.1.1.c.3. Region 4 is designed to accommodate fresh fuel up to 4.95 +/- 0.05 wt% U-235 initial enrichment, or spent fuel regardless of the discharge burnup in a 1-of-4 checkerboard arrangement of 1 fresh assembly with 3 spent Type T fuel assemblies with burnup and cooling times in accordance with Design Feature 5.6.1.1.c.4.

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 < 1.0 be evaluated in the absence of soluble boron. Hence, the design of all 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 (Ref. 5) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental mishandling of a fresh fuel assembly face adjacent to a fresh fuel assembly of Region 3. This could potentially increase the criticality of Region 3. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal conditions is 300 ppm and 700 ppm under the most severe postulated fuel mis-location accident. Safe operation of the spent fuel storage racks may therefore be achieved by controlling the location of each assembly in accordance with Design Features 5.6 FUEL STORAGE. During fuel movement, it is necessary to perform Surveillance Requirement 4.7.13.2.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of any one of the three regions. Examples of these accident conditions are the loss of cooling and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in all regions. The most limiting postulated accident with respect to the storage configurations assumed in the spent fuel rack

BASES

APPLICABLE SAFETY ANALYSES (continued)

criticality analysis is the misplacement of a nominal 4.95 ± 0.05 w/o U-235 fresh fuel assembly into an empty storage cell location in the Region 3 checkerboard storage arrangement. The amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 due to this fuel misload accident is 700 ppm (Ref. 1 and Ref. 9).

A spent fuel boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the k_{eff} design basis limit of 0.95 (Ref. 6). The spent fuel pool boron dilution analysis concluded that an inadvertent or unplanned event that would result in a dilution of the spent fuel pool boron concentration from 2000 ppm to 700 ppm is not a credible event.

The concentration of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The spent fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the spent fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 7. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

Action a:

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored along with suspending movement of fuel assemblies.

Action a is modified by a provision indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4 is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

4.7.13.1

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no significant replenishment of pool water is expected to take place over such a short period of time. (Ref. 6)

4.7.13.2

This Surveillance Requirement verifies that the concentration of boron in the spent fuel storage pool is within the required limit during fuel movement until the final configuration of the assemblies in the storage racks is verified to be correct. As long as this Surveillance Requirement is met, the analyzed accidents are fully addressed. The 72 hour Frequency provides additional assurance that the maximum k_{eff} remains below the 0.95 limit under the postulated accident condition. (Ref. 1, 8, and 9)

REFERENCES

1. Stanley E. Turner (Holtec International), "Criticality Safety Analyses of Sequoyah Spent Fuel Racks with Alternative Arrangements," HI-992349

2. B.K. Grimes (NRC GL78011), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978

3. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", August 19, 1998

4. UFSAR, Section 4.3.2.7, "Critically of Fuel Assemblies"

5. 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).

6. K K Niyogi (Holtec International), "Boron Dilution Analysis," HI-992302

7. FSAR, Section 15.4.5

8. NRC letter to TVA dated August 1, 1990, " Increase Fuel Enrichment to 5.0 Weight Percent (TAC Nos. 76074, 76075, 76774, 76775) (TS 90-12) - Sequoyah Nuclear Plant, Units 1 and 2"

9. Stanley E. Turner (Holtec International, "Evaluation of the Effect of the Use of Tritium Producing Burnable Absorber Rods (TPBARS) on Fuel Storage Requirements," HI-2012629

B 3.7 PLANT SYSTEMS

3/4.7.14 CASK PIT POOL MINIMUM BORON CONCENTRATION

BASES

This specification is deleted.

Pages B3/4 7-13 through B3/4 7-15 are deleted.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NO. 327, 328

FRAMATOME-ADVANCED NECLEAR POWER (ANP)
REPORT NO. BAW 10237, Revision 1

BAW-10237
September 2001
Revision 1

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

(Unclassified, Non-Proprietary Version)

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Lynchburg, Va. 24506-0935

SEQUOYAH TOPICAL REPORT
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LIST OF ACRONYMS

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

LIST OF ACRONYMS

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

LIST OF ACRONYMS

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

LIST OF ACRONYMS

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

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 (TPCTR) 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 TPCTR and the SER requiring the performance of plant specific evaluations and analyses to demonstrate that no significant safety issues are raised by the irradiation of TPBARs.

This report addresses the required plant specific evaluations and analyses completed for SQN to demonstrate that there are no significant safety or operational issues when TPBARs are incorporated into SQN core designs and plant operations. 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. LOCTAJR

    b. Anticipated Transients Without Scram (ATWS)

    Items 1.a and 1.b have been approved and closed in SERs dated January 17, 2001 and March 16, 2001 respectively.

2. Identifies and evaluates the significant differences as they apply to SQN relative to the TPCTR.

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

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

5. Addresses plant specific changes consisting of:

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

    b. SQN thermal power up-rate of 1.3%. The uprate is not required for the implementation and utilization of TPBARs, however, analyses and evaluations performed for this report assumed up-

rated thermal power conditions because TVA anticipates implementation of this uprate prior to initial insertion of TPBARs into SQN.

6. 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 a 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 SQN's operating licenses 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.

SECTION 1 INTRODUCTION

1.1 PURPOSE OF PROGRAM

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

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

1.2 DESCRIPTION OF EFFORT

This report addresses the required plant specific evaluations and analyses completed for SQN to demonstrate that there are no significant safety or operational issues when TPBARs are incorporated into SQN core designs and plant operations. 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. LOCTAJR

   b. Anticipated Transients Without Scram (ATWS)

   Items 1.a and 1.b have been approved and closed in SERs dated January 17, 2001 and March 16, 2001 respectively.

2. Identifies and evaluates the significant differences as they apply to SQN relative to the TPCTR.

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

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

5. Addresses plant specific changes consisting of:

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

   b. SQN thermal power up-rate of 1.3%. The uprate is not required for the implementation and utilization of TPBARs, however, analyses and evaluations performed for this report assumed up-rated thermal power conditions because TVA anticipates implementation of this uprate prior to initial insertion of TPBARs into SQN.

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

   a. TPBAR surveillance program.

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

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

1.3 SEQUOYAH PLANT PARAMETERS

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

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

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

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

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

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

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

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

1.4.1 Sequoyah Report Sections Referencing the TPC Topical Report

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

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

1.4.2 Identification of Differences

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

1.5 SEQUOYAH PLANT SPECIFIC INTERFACE ISSUES

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

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

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

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

1.5.1 Handling of TPBARs

Action

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

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

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

Response

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

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

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

Consolidation Sequence

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

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

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

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

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

Subsequently, a shipping cask is placed into the cask loading pit. The cask is handled by the Auxiliary Building crane in accordance with NUREG-0612 program requirements. The canisters are transferred into the submerged cask. The cask is removed from the cask loading pit, drained of water and

decontaminated, packaged and certified for shipment. This shipping process is repeated until all TPBARs irradiated during the past operating cycle have been shipped. 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.

1.5.2 Procurement and Fabrication Issues

Action

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

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

Response

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

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

Company LLC that operates under a separate Board Of Directors. WesDyne uses the Westinghouse Quality Management System (QMS).

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

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

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

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

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

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

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

1.5.3 Compliance with DNB Criterion

Action

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

Response

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

References

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

1.5.4 Reactor Vessel Integrity Analysis

Action

NUREG-1672, Section 2.5.3, "The TPC topical report identifies the applicable regulations and describes methods for demonstrating compliance with Appendices G and H to 10 CFR Part 50 and with 10 CFR 50.61. In the TPC topical report, DOE concludes, and the staff agrees, that the reference plants pressure/temperature limits report (PTLR) and final safety analysis report (FSAR) would need to be updated to reflect the change to the PTS value and include the updated P-T curves for the applicable EFPYs. In addition, because the reactor vessel integrity analyses are dependent upon the plant-specific materials properties and neutron fluence, the staff concludes that a licensee participating in DOE's

program for the CLWR production of tritium must present the material properties for its reactor vessel and perform analyses that demonstrate it will meet the requirements of Appendices G and H to 10 CFR Part 50 and of 10 CFR 50.61. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

Response

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

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

Input Parameters and Assumptions

Inlet Temperature

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

Fluence Projections

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

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

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

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

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

Applicable Analyses

Surveillance Capsule Withdrawal Schedule

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

Heat-up and Cooldown Pressure - Temperature Limit Curves

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

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

Pressurized Thermal Shock (PTS)

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

Emergency Response Guideline (ERG) Limits

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

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

Upper Shelf Energy (USE)

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

Conclusions

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

1. The core design employs power suppression techniques which may include the insertion of BPRAs in key peripheral fuel assembly locations so that the power in those locations remains comparable to that in the current Sequoyah loading patterns.

2. The inlet temperature for the tritium production core remains within the range of validity for the RV integrity analysis equations.
3. The fluence projections for the tritium production core are bounded by the existing fluence projections for SQN. Therefore, the existing RV integrity analyses remain valid for the Tritium Program.

References

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

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

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

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

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

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

1.5.5 Control Room Habitability Systems

Action

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

Response

The acceptance criteria for habitability of the Main Control Room following a design basis accident are based on meeting the relevant requirements of General Design Criteria (GDC) 4, 5, and 19 of 10 CFR Part 50 Appendix A. The documented design basis for the Sequoyah Nuclear Plant Main Control Room systems provides adequate protection of Control Room personnel for operation with a conventional (non-tritium producing) core. The NRC in the SER written for the DOE Topical Report on the reference plant concurred that only the radiation dose criteria are potentially affected by the incorporation of the TPBARs. The NRC noted that the major habitability concern for the referenced plant was the direct consequence

of the assumed high leak rate from the Emergency Core Cooling System (ECCS). The 2 gpm assumed leak rate is the value formerly used as a default for plants without a leakage reduction system. The ECCS leakage normally assumed in accident assessments is twice the leak rate that triggers corrective action under the applicable leak reduction program. The NRC further noted that values of 2 gallons per hour or less which are typically used would meet the relevant dose criterion.

An analysis was performed for Sequoyah 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 Guidance Reports (References 1.5.5.1 and 1.5.5.2) 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.

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.

References

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

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

1.5.6 Specific Assessment of Hydrogen Source and Timing of Recombiner Operation

Action

NUREG-1672, Section 2.9.2, "The staff agrees with the DOE conclusions, based on the conservative assessment of the TPBARs on the combustible gas concentrations in containment following a LOCA, that the combustible gas control systems are not expected to be affected by the TPC. However, the staff concludes that a plant-specific assessment is required to quantify the sources and to determine the time at which initiation of recombiner operation should commence to limit the hydrogen concentration to acceptable levels. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

Response

Introduction

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

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

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

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

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

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

TPBAR Metal-Water Reaction

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

However, if the TPBAR cladding is breached following a LBLOCA, the potential for a metal water reaction with internal zirconium components can be postulated.

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

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

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

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

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

TPBAR Tritium and Hydrogen Inventories

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

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

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

Results and Conclusions

The additional hydrogen inventories that are conservatively estimated to be associated with a TPC are 1,122 scf associated with zirconium-water reactions with the TPBAR getter materials, 357 scf of tritium

gas from the TPBARs, and 16 scf of hydrogen from $^3He(n,p)T$ reactions inside the rods. This sums to a total of 1,495 scf as the potential additional amount of hydrogen contributed by the TPBARs following a LBLOCA.

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

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

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

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

References

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

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

1.5.7 Light – Load Handling System

Action

NUREG-1672, Section 2.9.1, "DOE evaluated the effect of TPBARs on the light load handling system for the reference plant against the guidance of SRP Section 9.1.4. DOE states, and the staff agrees, that the incorporation of the TPBARs has no effect on this system. However, DOE concludes, and the staff agrees, that because of the increase in weight of TPBARs compared to burnable poison rod assemblies, this effect should be evaluated on a plant-specific basis. The staff has identified this as an interface item

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

Response

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

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

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

1. In accordance with NUREG-0612, -0554 and ANSI N14.6, the Spent Fuel Pit Bridge crane and canister lifting device will contain sufficient aspects of the single failure proof criteria to preclude a drop of the loaded canister as delineated below.
   a) The SFP Bridge crane is considered equivalent-single-failure proof with respect to structural integrity in accordance with NUREG-0612 (NUREG-0554) due to the following:
      1) Since the SFP Bridge crane has a capacity of 2000 lbs. and the weight of the submerged loaded canister is approximately 700 lbs., the crane has safety factors twice the normally required values.
      2) The crane is equipped with redundant high hook limit switches of different designs to preclude structural failure.

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

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

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

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

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

1.5.8 Station Service Water System

Action

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

Response

Introduction

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

Tritium Impact on Spent Fuel Pool Decay Heat

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

Increased Spent Fuel Pool Cooling Heat Rejection on ERCW

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

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

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

Conclusions

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

1.5.9 Ultimate Heat Sink

<u>Action</u>

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

Response

Introduction

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

Tritium Impact on Spent Fuel Pool Decay Heat

See previous discussion under Interface Item 1.5.8.

Increased Spent Fuel Pool Cooling Heat Rejection on UHS

The design basis analysis for the UHS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant temperature increase in the UHS. The increase in decay heat associated with TPC is approximately 1.7 MBTU/Hr. The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 8 MBTU/Hr. This total increase in decay heat load is well within the design bases limiting heat load imposed on the ERCW and UHS during other modes of operation. Increased ERCW flows are the same higher flow rates that have been specified during other modes of operation. This small amount of increased decay heat and increased ERCW flow, when compared to the overall flow rates of the UHS through the ERCW System, produces an insignificant increase (< 0.1°F) in UHS temperature leaving the plant site. Since there is no significant increase, and since the ERCW has significant margin available, no changes to the ERCW temperature requirements are warranted.

Conclusions

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

1.5.10 New and Spent Fuel Storage

Action

NUREG-1672, Section 2.9.2, "The staff reviewed the effect of storing fuel assemblies with TPBAR assemblies in the new and spent fuel racks for the reference plant in accordance with SRP Section 9.1.1

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

Response

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 U^{235} can be stored in the fresh fuel rack array utilizing 146 specific cells of the 180 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 spent fuel storage racks. This reanalysis was performed with fuel assemblies of nominal enrichments up to 5.0 w/o U^{235} containing TPBARs and also addressed other neutron poisons including Burnable Poison Rod Assemblies (BPRA) and Gadolinia integral absorbers rods. The fuel was assumed to operate with TPBARs or BPRAs, which were removed at the time the assemblies were placed in storage. As in the current analysis, credit was taken for soluble boron, fuel burnup, and cooling times, where appropriate.

The reanalysis adequately accounted for the effects of operating with TPBARs and confirmed that Technical Specification changes were required. Burnup vs cooling time curves, applicable to fuel burned with TPBARs, will be added to the Technical Specifications. No change is required in the checkerboard storage patterns or the amount of soluble boron providing the 5% margin to criticality.

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 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

Action

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

Response

Introduction

The SFPCCS for SQN is sized to handle full core off-loads. In the 1994-95 timeframe, SQN underwent spent fuel storage rack additions, which included development of a new thermal hydraulic analysis based on standard NRC approved methodologies which are scenario based. After the rerack design change TVA recognized the impracticality of following a scenario based set of limits during plant operation for predicting SFP decay heat load. Following the licensing efforts associated with the rerack modification at SQN, the FSAR was revised to capture a limiting value of decay heat that could be placed in the SFP, based on outage specific decay heat analysis performed for each outage. This approach provided a more realistic means (based on quantitative limits instead of scenario based limits) of assuring compliance with the maximum allowable design basis decay heat loads that could be placed in the SFP at any time. Compliance with these limiting values provides assurance that, should a train of SFPCCS

fail, maximum analyzed temperatures of the SFP and attendant decay heat removal system piping will not be exceeded.

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

Tritium Impact on SFP Decay Heat

See previous discussion under Interface Item 1.5.8.

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

Alternate SFP Decay Heat Analysis

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

SFP Heat Exchanger Fouling Factor

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

Component Cooling System Maximum Water Temperature

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

Results of Alternate Analysis

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

specific SFP design values and parameters for both the existing design and the proposed alternate design.

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

Impact of Higher Allowable Decay Heat in the SFP

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

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

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

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

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

Conclusions

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

1.5.12 Component Cooling Water System

Action

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

Response

Introduction

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

SER Section 2.9.4 indicated that the primary concern of the TPC impact on CCS was the additional heat load imposed by the SFPCCS on CCS, and any required changes to flow to meet the increased heat removal demand. The SER also indicated that if the impact on CCS was significant, the ability of the

CCS to serve other safety related heat exchangers (e.g. Residual Heat Removal System (RHRS)) may be affected.

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

Tritium Impact on Spent Fuel Pool Decay Heat

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

Increased Spent Fuel Pool Cooling Heat Rejection on CCS

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

CCS design thermal analyses have been revised to reflect increased SFPCCS allowable decay heat loads. CCS flows to the SFPCCS heat exchangers have not been increased. The additional heat load rejected to the CCS from the SFPCCS heat exchanger results in slightly elevated CCS temperatures, but are well within existing design basis values. Piping analysis and support analysis of the CCS have been previously analyzed at a higher ultimate temperature associated with more bounding operational modes, and are not affected by the increased CCS heat load. The mixing of multiple CCS return lines into

common headers minimizes the impact of the elevated CCS temperatures, since as SFPCCS heat loads increase, the RHRS heat loads decrease. With all CCS flows returning to a common header prior to returning to the CCS/ERCW heat exchangers, there is no measurable change to the mixed stream CCS temperature.

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

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

Conclusions

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

1.5.13 Demineralized Water Makeup System

Action

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

Response

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

requires additional demineralized water as makeup. The SER required the specific impact on DWMS from increased feed and bleed demand be evaluated.

TVA does not intend changes to the plant's current feed and bleed operations to control boron concentration in the RCS. Continuation of the current feed and bleed program will result in the RCS observed maximum tritium levels of 2.5 μCi/gm increasing to around 9 μCi/gm with the TPC. This increase is due to normal reactor tritium production plus the tritium permeation from TPBARs. 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 μ Ci/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 SQN DWMS there exists sufficient surge capacity as well as production capacity to meet these projected needs. As tritium levels increase in the RCS, ample planning time will be available to assure adequate surge volume is available and production rates are capable of meeting demand.

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

Conclusions

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

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

1.5.14 Liquid Waste Management System

Action

NUREG-1672, Section 2.11.2, "On the basis of the preceding discussion, the staff concludes that in both cases (the design-basis TPBAR permeation of tritium and the failure of two TPBARs) there is a sufficient

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

Response

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

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

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

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

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

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

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 Offsite Dose Calculation Manual (ODCM) release limits.

Conclusions

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

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

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.

1.5.15 Process and Effluent Radiological Monitoring and Sampling Systems

Action

NUREG-1672, Section 2.11.5, "In Section 2.11.6 of the TPCTR, DOE states that the current process and effluent radiological monitoring instrumentation and sampling systems that are in place at the reference plant, as well as at other operating PWR plants, include the capability for monitoring the tritium levels within the plant and in plant effluent pathways, and are adequate for use when the plant is operated with a TPC. On the basis of its review, the staff agrees with DOE that the existing capability for radiation monitoring is adequate for tritium levels at the reference plant. In response to the staff's RAI dated

October 15, 1998, DOE stated that the details of the laboratory instrumentation and sampling frequencies and locations are plant dependent. Therefore, a plant-specific assessment of the candidate plant for the TPC will be required to provide such information. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium."

<u>Response</u>

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

Tritium Monitoring

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

Air Sampling

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

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

Liquid Monitoring

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

Liquid Scintillation Counting

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

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

Conclusions

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

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

1.5.16 Use of LOCTA-JR Code for LOCA Analyses

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

Response

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

References

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

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

1.5.16.3 Letter from NRC (Robert E. Martin) to TVA (J.A. Scalice) dated January 17, 2001, regarding SAFETY EVALUATION OF LOCTAJR CODE FOR LOSS -OF-COOLANT ACCIDENT

ANALYSIS OF FUEL RODS - WATTS BAR NUCLEAR PLANT, UNIT 1, AND SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA9520, MA9583, MA9584).

1.5.17 ATWS Analysis

Action

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

Response

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

References

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

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

1.6 SEQUOYAH PLANT SPECIFIC CHANGES

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

1.6.1 Technical Specifications

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

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

1.6.2 Sequoyah Specific TS Changes

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

1. TS Table 3.3-9 – Remote Shutdown Monitoring Instrumentation – Revised Source Range Measurement Range
2. TS 3/4.5.1 – Cold Leg Accumulator – Boron Concentration Increase
3. TS 3/4.5.5 – Refueling Water Storage Tank – Boron Concentration Increase
4. TS 3/4.7.14 – Cask Pit Pool Minimum Boron Concentration – Deletion of Requirements for Storing Spent Fuel in the Cask Pit
5. TS 5.3 Design Features/Reactor Core/Fuel Assemblies – Limitation for TPBARs
6. TS 5.6 Design Features, Fuel Storage – Revised Storage Requirements for Fuel Assemblies Containing TPBARs

These TS changes and related TS Bases changes are further discussed in Enclosure 1 of the License Amendment Request (LAR). This submittal to the NRC will request an amendment to the SQN operating license to allow operation with a tritium production core. The NRC in their SER for the TPCTR identified several potential TS changes (see Section 1.6.1) that could be required to support operation with TPBARs. Two of the identified TS changes are not required for SQN. Their applicability to SQN is discussed below:

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

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

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

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

1.6.3 Thermal Power Uprate

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

1.7 REFERENCES

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

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

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

Table 1-1

NSSS Performance Parameters

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

Table 1-2

Core Design Parameters for the Sequoyah Tritium Production Cores

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

Table 1-3

Key Physical Parameters for Sequoyah Units

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

Table 1-4

Summary of Standard Review Plan (SRP) Evaluations

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

Table 1-4

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

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

Table 1-4

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

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

Table 1-4

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

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

Table 1-4

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

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

Table 1-4

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

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

Table 1-4

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

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

Table 1-4

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

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

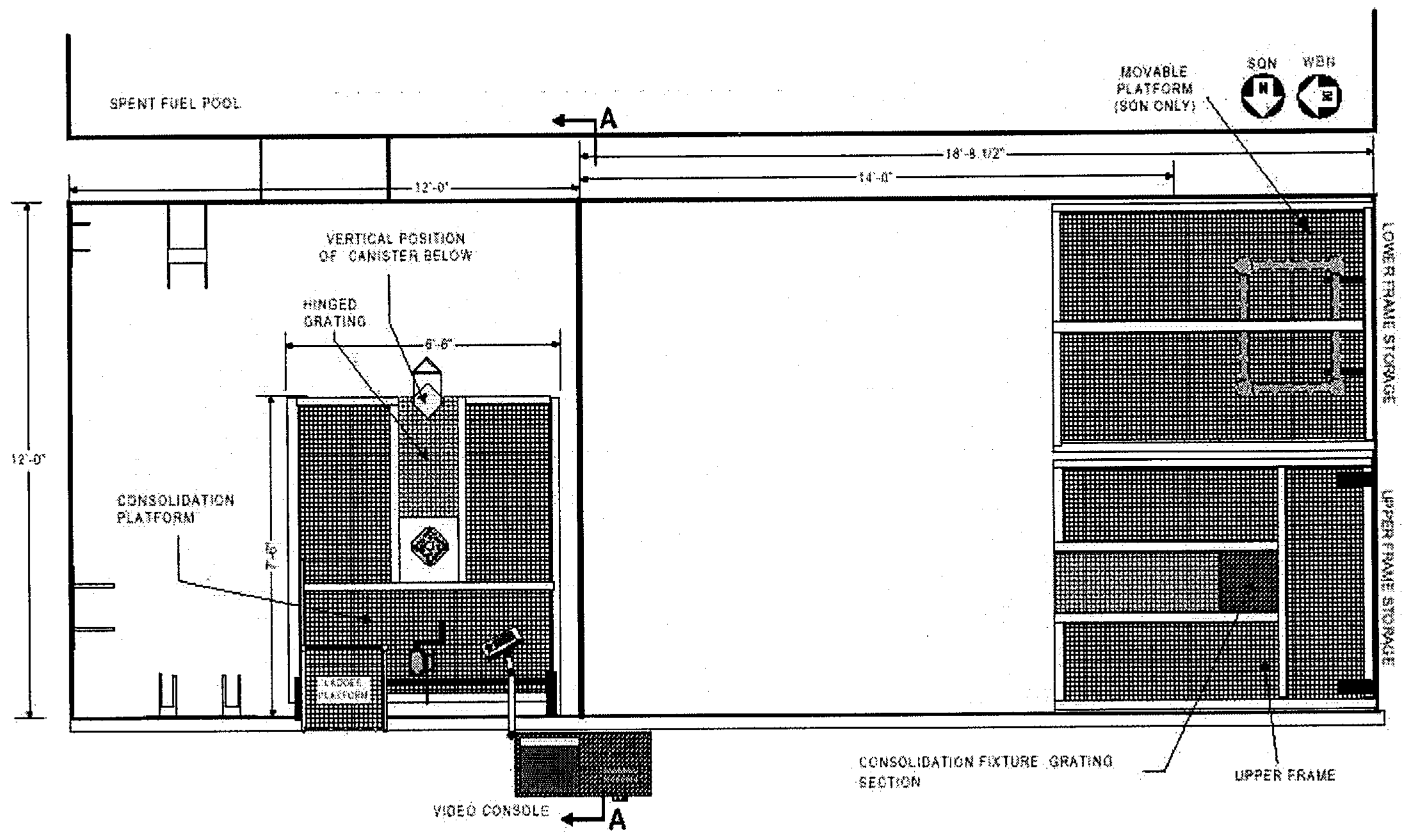


Figure 1.5.1-1

Consolidation Plan

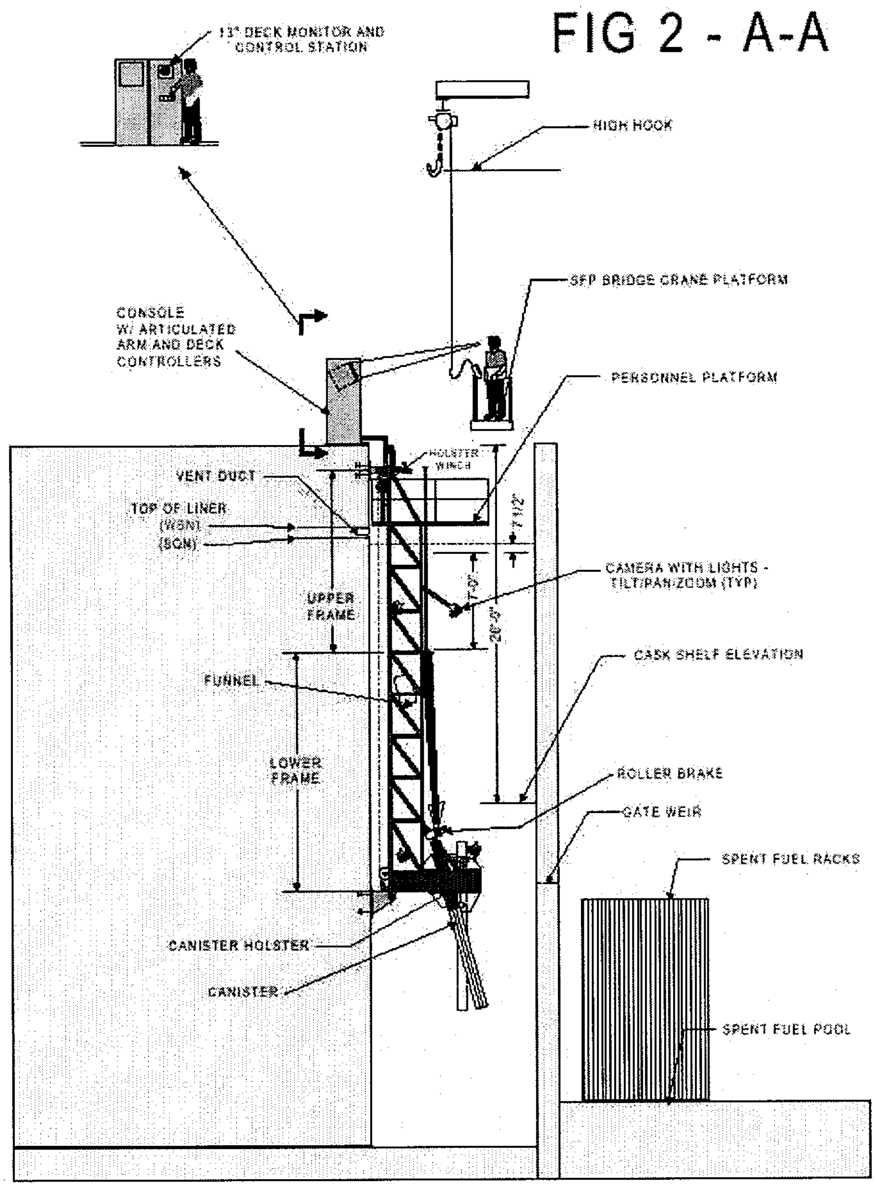
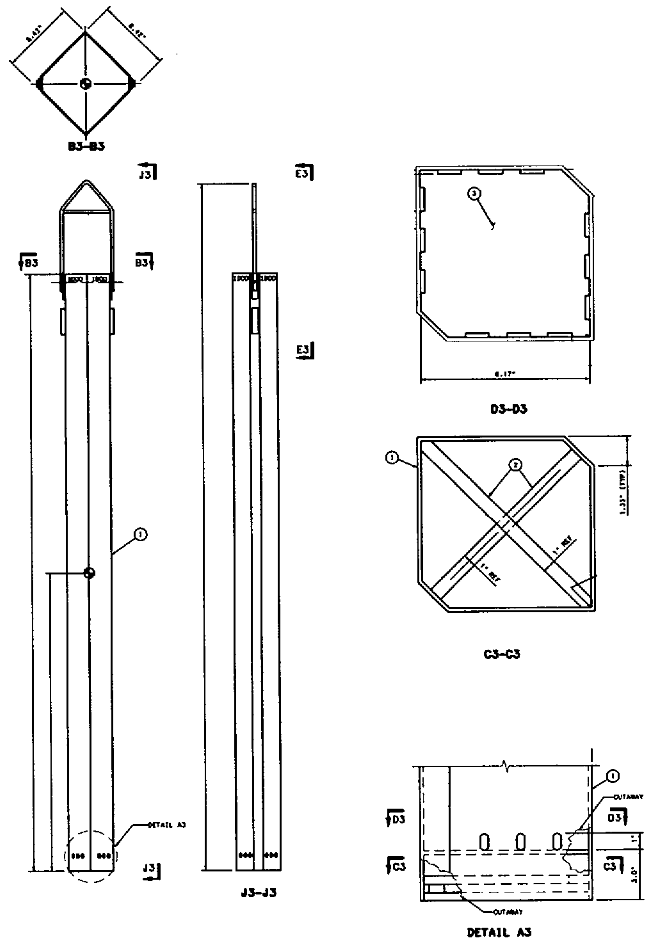
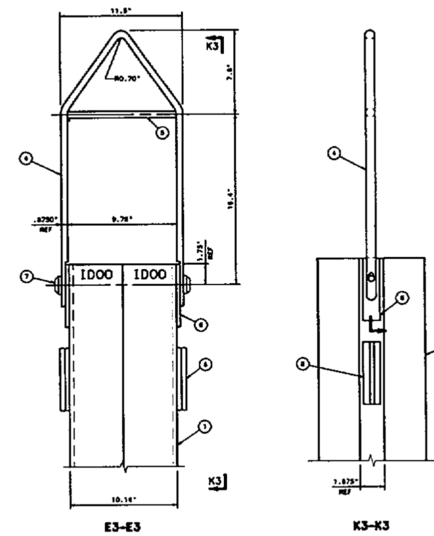


Figure 1.5.1-2

Consolidation Plan A-A





1. EACH CANISTER SHALL HAVE A UNIQUE IDENTIFICATION CLEARLY DENOTED ON FOUR LOCATIONS OF THE EXTERIOR OF CANISTER AT TOP AS SHOWN (ID00). LETTERING IS TO BE MINIMUM ¾" TALL ETCHED, ENGRAVED OR STAMPED ON METAL.

2. TOTAL DRY WEIGHT: 904 LB (INCLUDING 300 TP BARS). DRY WEIGHT EMPTY: 184 LB.

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

Figure 1.5.1-3

Consolidation Canister

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

Figure 1.5.4-1

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