TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

John A. Scalice
Site Vice President, Watts Bar Nuclear Plant

FEB 10 1997

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

Gentlemen:

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

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING REQUEST FOR LICENSE AMENDMENT TO TECHNICAL SPECIFICATIONS - SPENT FUEL POOL STORAGE CAPACITY INCREASE (TAC NO. M96930)

The purpose of this letter is to provide TVA's response to the request for additional information which was attached to the January 14, 1997 meeting summary dated January 16, 1997. Enclosure 1 provides the response to the NRC's questions and Enclosure 2 documents information discussed in the January 14, 1997 meeting.

Enclosure 3 provides the commitment list from this letter. If you should have any questions, please contact P. L. Pace at (423) 365-1824.

Sincerely,

J. A. Scalice

J. A. Scalice

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Enclosures
cc: See page 2

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cc (Enclosures):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. Robert E. Martin, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
REQUEST FOR ADDITIONAL INFORMATION
SPENT FUEL POOL STORAGE CAPACITY

The following provides TVA's response to NRC's request for additional information attached to the January 14, 1997, meeting summary dated January 16, 1997.

PLANT SYSTEMS

QUESTION 1.

With regard to the calculated decay heat loads following the proposed pool expansion, provide the following information:

a. the decay heat generation rate as a function of decay time for both the maximum normal and maximum abnormal conditions (information should clearly show the decay heat generation rate from each batch of the previously discharged spent fuel assemblies and the freshly discharged full core in the spent fuel pool); and

b. a comparison of the assumptions used in the decay heat calculations to the guidance described in Section 9.1.3 of the Standard Review Plan (SRP, NUREG-0800) including the Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling."

RESPONSE 1

a. The decay heat generated by the previously stored assemblies in accordance with the discharge schedule assumed in Section 5.3 of the report attached to the license amendment request is provided in the attached Table 1 for ORIGEN2 and ASB 9-2.

The total decay heat load corresponding to the normal full core discharge (Case 1) and unplanned full core discharge (Case 2) can be found in Figures 5.5.6 and 5.5.8, respectively, of the report. Attached Figure 1 shows the total decay heat load plot using ORIGEN2 (duplicated from Figure 5.5.6) and ASB 9-2 procedures. It is noted that the ASB 9-2 procedures yield decay heat values which are approximately 27% greater than the more accurate ORIGEN2 results.

b. See the response to Question 2.

QUESTION 2

Identify and provide the basis for any deviations and exceptions to the guidance described in SRP Section 9.1.3 regarding the decay heat calculation and cooling of the spent fuel assemblies.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
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RESPONSE 2

The decay heat calculation and bulk pool temperature calculation in SRP 9.1.3 are based on a specified fuel inventory in the pool, on specific burnup of fuel, and on a postulated discharge scenario. The WBN decay heat load calculations use the actual fuel inventory in the pool at the point in time in the future when a full core off-load into the pool will exhaust its storage capacity. In other words, the heat load burden assumed to be imposed on the cooling system is much greater than the load which would be inferred from SRP 9.1.3.

For decay heat calculations, the SRP document refers to ASB 9-2 which is an exponential curve developed by the NRC some 20 years ago. Holtec International has used the ORNL developed code ORIGEN2 for decay heat load computation to realize greater accuracy in the decay heat load definition. A comparison of the decay heat load plot predicted by ORIGEN2 and ASB 9-2 for the normal full core off-load condition is provided in response to the previous question.

QUESTION 3

Figure 5.5.7 indicates that 600 hours after the reactor shutdown the decay heat generation rate in the spent fuel pool is approximately 25.5×10^6 Btu/Hr. However, Figure 5.5.8 indicates that 600 hours after the reactor shutdown the decay heat generation rate in the spent fuel pool is approximately 12.0×10^6 Btu/Hr. Provide detailed clarification for the discrepancy.

RESPONSE 3

There is not a discrepancy in the information presented because Figures 5.5.7 and 5.5.8 are for two different discharge scenarios. In Figure 5.5.7, the decay heat load in the spent fuel pool at 600 hours for the normal full core discharge case is due to 1,680 assemblies accumulated in the pool from the previous 21 discharges (reference Section 5.3) plus 193 assemblies discharged into the pool starting at 288 hours after reactor shutdown. This provides a total of 1873 assemblies. In the unplanned full core off-load shown in Figure 5.5.8, there are 1,600 assemblies from the previous 20 discharges plus one batch of 80 assemblies discharged into the pool starting at 288 hours after reactor shutdown. This provides a total of 1680 assemblies. Because of the smaller quantity of freshly discharged fuel resident in the pool at 600 hours for Figure 5.5.8, the total decay heat load is much less than the decay heat load for Figure 5.5.7.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
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SPENT FUEL POOL STORAGE CAPACITY

QUESTION 4

In the decay heat calculation for the case of normal full core discharge, fuel assemblies are assumed to be discharged to the spent fuel pool after 12 days of decay in the reactor. Has this restriction of 12-day duration for fuel assemblies to decay in the reactor prior to movement of a fuel assembly been incorporated in the Watts Bar Technical Specifications (TS)?

RESPONSE 4

A restriction of 12 days of decay duration has not been specified in the WBN TS. The 12 day in-core decay period is used in the bounding analysis to define the maximum heat load. Inasmuch as the thermal-hydraulic safety margins (peak fuel cladding temperature, local maximum pool water temperature and time-to-boil) are directly influenced by the peak decay heat load, it is more appropriate to predicate the discharge delay duration on heat loads. Restrictions will be incorporated into a plant operating instruction which reflect the following procedural process:

> Accurate decay heat loads for the core to be off-loaded and for any spent fuel existing in the pool will be determined based on the reactor operating history prior to each refueling in order to determine both the time at which off-loading will start and the rate of transfer into the spent fuel pool such that the maximum design spent fuel pool heat load of 32.60×10^6 Btu/hr is not exceeded. The in-core decay time must be the greater of 100 hours (as currently required by Technical Requirement (TR) 3.9.1) or the time required to ensure that the maximum design heat load limit will not be exceeded. For unplanned outages, a similar procedure shall be used. The in-core decay time shall be 288 hours (12 days) if a cycle-specific evaluation is not carried out.

The following paragraph should be added to Section 5.3 of the report attached to the license amendment request for clarification.

> The decay heat load analysis presented in this chapter is based on off-load scenarios which consider a 12 day delay time between shutdown and initiation of fuel transfer to the spent fuel pool. These scenarios also consider a concurrent maximum heat load in the pool resulting from the accumulation of spent nuclear fuel (SNF) in the spent fuel pool over many years of operation. In both the normal and emergency full core off-load conditions the inventory in the pool after discharge is considered to be 1873 assemblies, which slightly exceeds the actual capacity and is conservative when establishing the maximum heat load for each scenario. A 12 day delay time may be required, depending on the reactor operating history under these worst case conditions. However, unless the pool is full of SNF a considerable margin exists between the worst case heat loads documented in the

analyses and actual total heat load 12 days following shutdown. Decay heat loads for the core and from any SNF existing in the pool based on the reactor operating history are determined prior to each refueling in order to control both the time at which off loading starts and the rate of transfer into the spent fuel pool such that the maximum design heat load of 32.60 x 10^6 Btu/hr (see Table 5.5.1, Case 2A, net heat load plus evaporative losses) established in this chapter is not exceeded. Decay heat loads are determined with methodology which uses ANS Standard 5.1, "Decay Heat Power in Light Water Reactors," and Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," as its basis.

QUESTION 5

Note #1 for TS Section 3.9.4 states that the spent fuel pool transfer canal gate and the spent fuel pool cask pit gate may travel over assemblies in the spent fuel pool. Discuss the design features (such as interlocks or single-failure-proof crane design) to preclude these gates from dropping on spent fuel.

RESPONSE 5

The features utilized to preclude dropping the gates on spent fuel are the design of the lifting device, the design of the crane, and the use of written procedures for removal and installation of the gates. The designs of the lifting device and crane are discussed in TVA's letter July 28, 1993, response to NRC's Generic Letter 81-07 concerning NUREG-0612 - Control of Heavy Loads and approved by NRC in Supplemental Safety Evaluation Report (SSER) 13. That letter states that lifting devices for loads identified in the response comply with the intent of ASME/ANSI-B30.9, "Slings," or ASME/ANSI-N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500kg) or More," as appropriate. Spent fuel pool gates are identified in that response as load A2. The design of the lifting device for the spent fuel pool gates meets this intent since the device has a design factor of safety against yield of 38 and a design factor of safety against ultimate of 77. The design of the Auxiliary Building overhead crane complies with the guidelines of CMAA Specification 70, "Specification for Electrical Overhead Traveling Cranes," and ASME/ANSI B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)." Written procedures, Maintenance Instruction (MI) 79.004, "Removal and Installation of the Spent Fuel Pit Gate," and MI 0.045, "Control of Heavy Loads in Critical Lift Zones - NUREG-0612," address handling of the gate. Also, in order to move the gates, the crane interlocks which prevent movement of other heavy loads over the pool have to be overridden using limit switch bypass keys. The use of these keys is controlled, and special precautions are enforced when the keys are used. This means that movement of the gates is closely controlled. As discussed in the report attached to the license amendment request, the spent fuel racks have been evaluated for a drop of a gate and are adequate to prevent damage to the stored fuel.

It should be noted that Section 3.9.4 is not in the WBN TS. This section is in the WBN Technical Requirements Manual (TRM) Section TR 3.9.4.

RADIOLOGICAL PROTECTION

QUESTION

It is recommended that the licensee revise this section of the TS, Chapter 9, "Radiological Evaluation," Table 9.2. The core inventory for gap-release fractions should be 12 percent instead of 10 percent (normal burnup) in accordance with NUREG/CR-5009. The increase in I-131 is due to the burnup level of 60 GWD/T and the increase in fuel enrichment to 5.0 percent of U-235, whereas Regulatory Guide 1.25 assumes a release fraction of 0.10 (normal burnup). The licensee must recalculate his fuel handling accident dose for I-131 to address NUREG/CR-5009.

RESPONSE

The current doses calculated in the report attached to the license amendment request are conservative with respect to NUREG/CR-5009, therefore, TVA does not intend to recalculate the fuel handling accident dose. Page 3.11, Paragraph 2 of NUREG/CR-5009 states that when compared with Regulatory Guide (RG) 1.25 requirements, the release fractions for I-131 are higher (12%) for extended burnup and higher enrichment, but that all other isotopic release fractions are lower. The last paragraph on that page concludes that RG 1.25 procedures can, therefore, be used for the extended burnup fuel.

Dose comparisons between RG 1.25 and NUREG/CR-5009 have been made and the NUREG/CR-5009 procedures result in lower total dose. The NUREG/CR-5009 iodine doses are slightly increased. However, the gamma and beta doses are significantly decreased. The result is a lower total dose.

TABLE 1

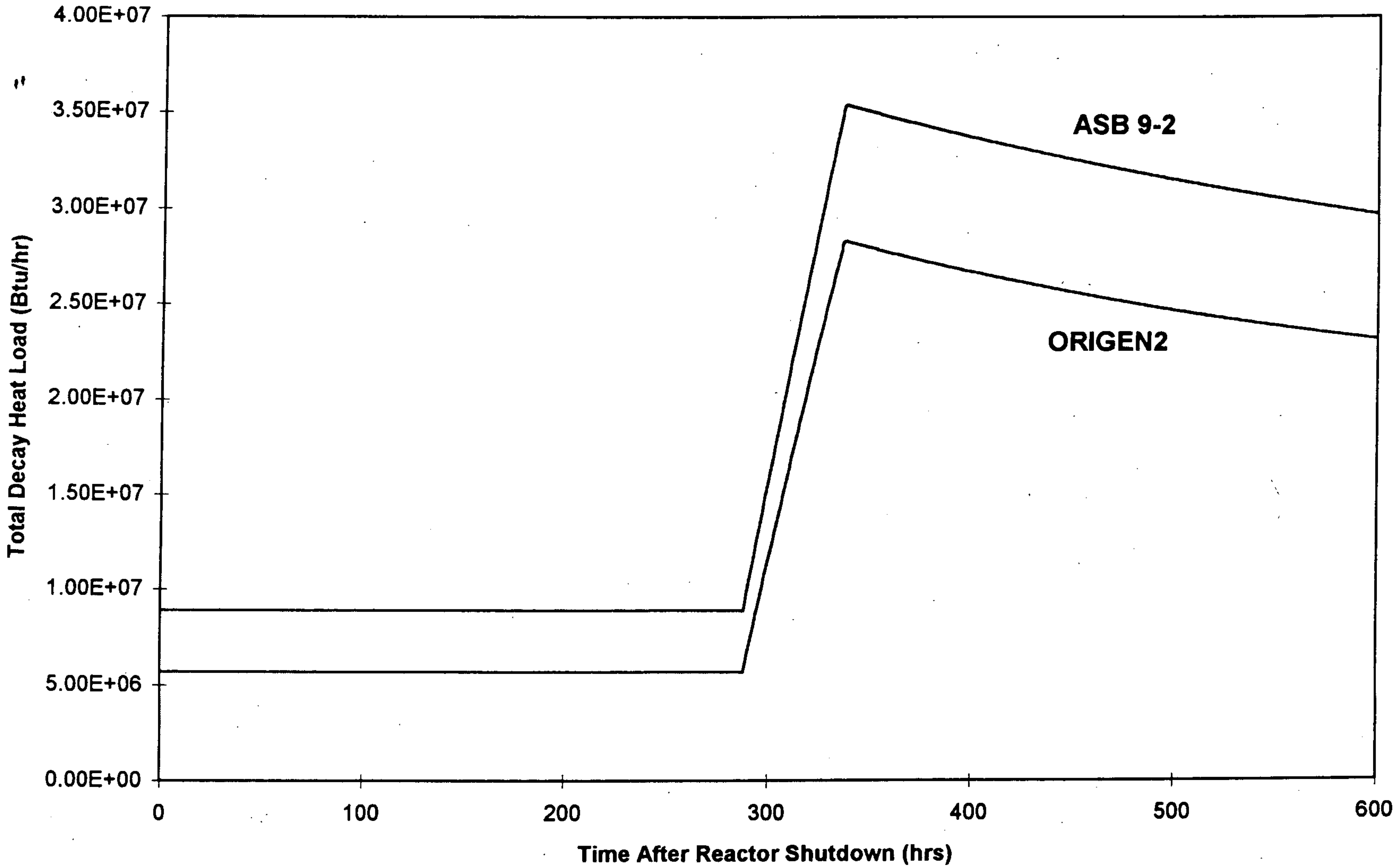
DECAY HEAT FOR PREVIOUSLY DISCHARGED FUEL BATCHES

Decay Heat Calculations Comparison
ORIGEN2 and ASB 9-2

Discharge Batch Number	ORIGEN2 Discharge Batch Decay Heat (Btu/hr)	ASB 9-2 Discharge Batch Decay Heat (Btu/hr)
1	140020.4	263598.4
2	143804.7	273249.6
3	147715.2	282900.8
4	151878.0	293758.4
5	156166.9	304012.8
6	160581.9	315473.6
7	165249.3	326934.4
8	170295.0	338998.4
9	175467.0	351062.4
10	181143.5	363729.6
11	187324.6	377000.0
12	194010.2	390873.6
13	201831.2	405350.4
14	211039.7	419827.2
15	222771.2	435510.4
16	238791.5	451796.8
17	263263.6	469892.8
18	306152.7	493417.6
19	394706.1	536848.0
20	611422.3	659297.6
21	1244794.0	1132809.6

FIGURE 1

Total Decay Heat Profiles for ASB 9-2 and ORIGEN2
(Full Core Discharge, Previous Inventory = 1680 assys)



ENCLOSURE 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
ADDITIONAL CONCERNS FROM JANUARY 14, 1997
SPENT FUEL POOL STORAGE CAPACITY

The following concerns were addressed in the January 14, 1997, meeting in Rockville, MD and are being documented in this letter.

QUESTION 1

What is the maintenance schedule for the spent fuel pool cooling heat exchangers?

RESPONSE 1

Under the WBN Preventive Maintenance Program, the spent fuel pool cooling heat exchangers are inspected during the second refueling outage, then every fourth refueling outage thereafter, if no abnormal indications are observed. Inspections are performed in accordance with Maintenance Instruction 78.05, "Inspection and Repair of Tubes in the Spent Fuel Pit Heat Exchanger."

QUESTION 2

Explain why it is acceptable to change the pool temperature from 150 degrees F to 159 degrees F.

RESPONSE 2

Acceptability of Increased Maximum Spent Fuel Pool Water Temperature

New spent fuel pool temperatures are deemed acceptable based upon the specifics of the WBN spent fuel pool cooling system; the margins and conservative assumptions utilized in the WBN analysis; the capability of WBN to cope with single failures in the cooling system and a total loss of cooling; and the margins which exist between maximum bulk and local pool temperatures and the pool boiling temperatures.

The present calculation of record for WBN determines a maximum spent fuel pool water temperature of 150 degrees F concurrent with the maximum design heat load of 26.272 x 10^6 Btu/hr. This maximum temperature/maximum heat load condition is based on single spent fuel pool cooling and cleaning system (SFPCCS) heat exchanger operation, emergency full core off-load after a 60 day decay time in the core, and decay heat loads from spent nuclear fuel (SNF) accumulated in the spent fuel pool over many years of two unit operation.

The new analysis predicts a maximum temperature of 159.24 degrees F and a maximum total heat load (net heat load plus coincident evaporative heat loss) of 32.6 x 10^6 Btu/hr. These new maximums are also based on single heat exchanger operation, unplanned full core discharge, and decay heat from existing SNF accumulated over many years of one unit operation.

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
ADDITIONAL CONCERNS FROM JANUARY 14, 1997
SPENT FUEL POOL STORAGE CAPACITY

Margins and Conservatism Included in the New Thermal Hydraulic Analyses

SFPCCS Heat Exchangers

Version 5.03 of the computer program STER was used in the new analysis to evaluate the performance of the SFPCCS heat exchangers with varying coolant water temperatures. Conservative assumptions used in development of the model include the following:

- The overall heat transfer coefficient used in the analysis was 351 Btu/(hr-ft^2-°F) per heat exchanger which is approximately 62% of the clean coefficient of 565 Btu/(hr-ft^2-°F).
- 5% of the tubes were considered plugged.
- No credit for heat transfer in the U-bend region of the tubes since this region is typically not in cross flow which reduces the heat transfer efficiency in that region. Not crediting the surface area in this region conservatively minimizes the calculated heat transfer. Spent fuel pool heat exchangers are assumed to continually be supplied with 95 degrees F component cooling system (CCS) water. Also, the CCS heat exchangers are assumed to be continually supplied with essential raw cooling water (ERCW) cooling water based upon a river water temperature ≤85 degrees F (the current TS Section 3.7.9 limit).

SFPCCS Pumps

The increased spent fuel pool temperature of 159.24 degrees F is well below the maximum allowable spent fuel pool temperature which would result in inadequate net positive suction head (NPSH) of 190 degrees F concurrent with minimum spent fuel pool water level. Other important parameters are listed in the table below.

PARAMETER	ORIGINAL VALUE	NEW VALUE
TEMPERATURE (°F)	150	~159
BHP @ 2300gpm (hp)	79.9	82.8
BHP @ runout point (hp)	86.3	87.2
MOTOR RATED (hp)	100	100

Additionally, WBN has 3 spent fuel pool pumps which are powered from diesel backed power supplies available. One pump is required to maintain pool temperature at ~159 degrees F.

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
ADDITIONAL CONCERNS FROM JANUARY 14, 1997
SPENT FUEL POOL STORAGE CAPACITY

Decay Heat Load

For both the normal full core and unplanned discharge scenarios, a total spent fuel pool inventory of 1,873 assemblies is considered in the analysis. This exceeds the actual capacity by 38 assemblies and thus slightly over estimates the total heat load. The U-235 enrichment of the fuel assemblies in the new analyses is conservatively assumed to be 3.5% although higher enrichments are to be used in future refueling cycles. When compared with fission of higher enrichment fuel at an equal burnup, the altered power spectrum of the lower enrichment fuel results in a slightly higher decay heat generation rate.

Time to Boil

Although initial temperatures are greater in the new analysis, sufficient time exists to establish makeup water from a qualified source. Addition of 100 degrees F water at 55 gpm within 10 hours following loss of cooling results in the spent fuel pool water level dropping to approximately 21 feet above the top of the racks, much greater than the required minimum of 10 feet. The following table compares original and new parameters pertinent to time to boil calculations.

PARAMETER	ORIGINAL VALUE	NEW VALUE
NORMAL REFUELING - 2 HX STARTING TEMP (°F) STARTING HEAT LOAD (Btu/hr) TIME TO BOIL (hrs)	 120 23.772×10^6 9.74	 124.69 28.10×10^6(NET + EVAP) 8.84
NORMAL REFUELING - 1 HX STARTING TEMP (°F) STARTING HEAT LOAD (Btu/hr) TIME TO BOIL (hrs)	 150 23.772×10^6 6.56	 151.17 27.91×10^6(NET + EVAP) 6.27
UNPLANNED DISCHARGE - 2HX STARTING TEMP (°F) STARTING HEAT LOAD (Btu/hr) TIME TO BOIL (hrs)	 120 26.272×10^6 8.81	 129.3 32.6×10^6(NET + EVAP) 8.99
UNPLANNED DISCHARGE - 1 HX STARTING TEMP (°F) STARTING HEAT LOAD (Btu/hr) TIME TO BOIL (hrs)	 150 26.272×10^6 5.94	 159.24 32.42×10^6(NET + EVAP) 5.86

Transient Temperature Response Determination

In addition to the conservatism built into the decay heat load calculation discussed above, the thermal capacity of the spent fuel pool is based only on the water volume. The metal fuel racks possess an appreciable thermal capacity. Ignoring this capacity, the thermal capacity was conservatively reduced. Heat conduction from the pool through the pool walls and floor slab has also been conservatively neglected.

Maximum Local Temperature Calculations

The spent fuel pool water elevation is assumed to be at the low water level elevation which results in a minimized local saturation temperature at the top of the racks (approximately 240 degrees F). The radial peaking factor is applied to one-third of the discharged core, and these hotter-than-average assemblies are placed together in the spent fuel pool. Assemblies in the discharged core generate heat at different rates as a result of their respective positions in the reactor. Applying the radial peaking factor results in a localized region with extreme heat generation, and correspondingly bounding local temperatures. Baby rack cell dimensions are used for the storage locations in the spent fuel pool. The pitch of the baby rack cells is smaller than for the other racks in the pool, resulting in a smaller available area for flow. The smaller area results in the highest resistance and, consequently, the lowest velocities and highest temperatures. The cask pit floor elevation was modeled to be at the same elevation as the spent fuel pool floor elevation. Since the cask pit floor elevation is actually lower than the spent fuel pool floor, and any rack in the cask pit must be elevated to allow use of the bridge crane, a large flow plenum results which is conservatively ignored. The fuel with the highest generation rate is assumed to be located near the cask pit. Placing the hottest fuel near the cask pit, which has no forced cooling, conservatively maximizes the cask pit temperatures. Minimum rack-to-floor and rack-to-wall gaps were used which maximizes the hydraulic resistance of the bottom plenum and downcomers, thereby conservatively reducing the water velocities.

Based on the above, the increase in spent fuel pool temperature is judged to be acceptable and conservatively calculated.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
REQUEST FOR ADDITIONAL INFORMATION
SPENT FUEL POOL STORAGE CAPACITY
COMMITMENT LIST

A plant operating instruction will be revised to require restrictions to ensure that the maximum design heat load limits will not be exceeded. (See Response 4 to RAI)