

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Watertford, CT 06385



JUL 15 2002

Docket No. 50-336
B18684

RE: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 2
Response to a Request for Additional Information
Technical Specifications Change Request (TSCR) 2-10-01
Fuel Pool Requirements

In a letter dated November 6, 2001,⁽¹⁾ Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request in the form of changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes would: (1) increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core; (2) allow fuel to be located in 40 Region B Storage Cells, which are currently empty and blocked, and (3) credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$. Additionally, as a result of a subsequent discussion with the staff of the Nuclear Regulatory Commission (NRC), DNC provided a revision to the Significant Hazards Consideration (SHC) discussion (Attachment 1) in a letter dated December 27, 2001.⁽²⁾ The revised SHC discussion did not affect the conclusion of the safety summary and the original

⁽¹⁾ J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements," dated November 6, 2001.

⁽²⁾ J. A. Price letter to the U.S. NRC, "Millstone Nuclear Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements, Revised Significant Hazards Consideration Discussion," dated December 27, 2001.

Pool

SHC determination. In a letter dated June 26, 2002,⁽³⁾ a Request for Additional Information (RAI) was received from the NRC which contains thirteen (13) questions related to the aforementioned license amendment request.

Attachment 1 provides the DNC response to the June 26, 2002, RAI. This additional information provided in this letter will not affect the conclusions of the Safety Summary and SHC discussion in the DNC November 6, 2001, and December 27, 2001, letters.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

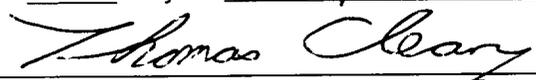
DOMINION NUCLEAR CONNECTICUT, INC.



J. Alan Price
Site Vice President - Millstone

Sworn to and subscribed before me

this 15TH day of JULY, 2002



Notary Public

My Commission expires FEB. 28, 2006

Attachment

cc: H. J. Miller, Region I Administrator
R. B. Ennis, NRC Senior Project Manager, Millstone Unit No. 2
NRC Senior Resident Inspector, Millstone Unit No. 2

⁽³⁾ R. Ennis (NRC) letter to J. A. Price, "Request For Additional Information, Spent Fuel Pool Requirements, Millstone Nuclear Power Station, Unit No. 2 (TAC No. MB3386)," dated June 26, 2002.

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Attachment 1

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Question 1

Your submittal stated that the boraflex in-service testing program consists of three parts. The first part is a direct examination of the boraflex material. The second part is blackness testing, and the third part is spent fuel pool silica monitoring. The staff requests the licensee to provide a discussion of the following questions pertaining to the licensee's assessment of the boraflex material condition in Regions A and B:

- a. What method does the licensee use to choose the panels to inspect? For instance, does the licensee perform Boron-10 Areal Density Gage for Evaluating Racks (BADGER) testing?

⁽¹⁾ J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements," dated November 6, 2001.

⁽²⁾ J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements, Revised Significant Hazards Consideration Discussion," dated December 27, 2001.

⁽³⁾ R. Ennis (NRC) letter to J. A. Price, "Request For Additional Information, Spent Fuel Pool Requirements, Millstone Nuclear Power Station, Unit No. 2 (TAC No. MB3386)," dated June 26, 2002.

- b. Does the licensee have plans to perform additional blackness testing prior to and after adding spent fuel to Regions A and B?
- c. The licensee's submittal states that spent fuel pool silica concentrations are measured and monitored for any unusual trend. Using the silica concentration can help determine the average boraflex loss but will not identify the most degraded panel. What is the licensee's method of accurately determining boraflex degradation (e.g., BADGER testing)?
- d. What are the licensee's current projections for boraflex degradation in Regions A and B? Also, does the licensee use RACKLIFE to make these projections?
- e. The licensee stated that the criticality analysis used 0.025 grams B-10/cm² (25% of the original design density) as the boraflex density. What are the actual boraflex density values in Regions A and B and how do these values compare to the actual boraflex density values in the spent fuel pool?

Response

- 1a. A summary of the Millstone Unit No. 2 boraflex in-service testing program is described in our submittal dated November 6, 2001, Attachment 1, pages 16 to 18. As described in our submittal, one part of our in-service boraflex testing involves removal of a boraflex poison box for destructive examination at specified time intervals. Our method for selecting which poison box is to be removed from the spent fuel pool storage racks is to use our Millstone Unit No. 2 RACKLIFE model to determine which boraflex poison box is at (or near) highest in predicted loss of boraflex. BADGER testing has not been performed at Millstone Unit No. 2. This is discussed further in the response to question 1c. below.
- 1b. As described in our submittal (Attachment 1, page 17), Millstone Unit No. 2 has already performed 3 boraflex blackness test programs. We have previously stated to the NRC in a letter dated October 24, 1996,⁽⁴⁾ (Attachment 2 page 8), that Millstone Unit No. 2 boraflex blackness testing "will continue to be performed periodically in the future, as long as Boraflex is credited in the criticality analysis." Our next blackness test program is planned to occur when the maximum accumulated gamma dose reaches an approximate value of 3.5E10 rads. Our most recent estimate of maximum gamma dose to the Millstone Unit No. 2 Boraflex is about 2.3E10 rads. Depending on exactly how gamma dose is accumulated, it will take several years to reach a gamma dose of 3.5E10 rads. The changes proposed in this license amendment do not alter in any significant

⁽⁴⁾ T. C. Feigenbaum to NRC, "Haddam Neck Plant, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Response to NRC Generic Letter 96-04, Boraflex Degradation in Spent Fuel Pool Storage Racks," dated October 24, 1996.

way the rate or distribution of gamma dose accumulation to the boraflex panels. Therefore our decision on when to perform blackness testing next is not affected by these proposed changes.

Summarizing our blackness test results to date, Millstone Unit No. 2 has tested about 50% of all the boraflex panels in the spent fuel pool at one time or another during the 3 blackness test campaigns, with many panels being tested in 2 or all 3 of the blackness test campaigns. A summary of the most recent blackness test results was described in our submittal dated November 6, 2001, Attachment 1, page 17. Also, see the response to question 11 for a more detailed discussion concerning the difference between the current boraflex gap model and the proposed boraflex gap model. Blackness test results to date have shown that boraflex gap formation is well within the EPRI predictions for boraflex gap formation, and well within the Millstone Unit No. 2 criticality analysis limits for gap size and distribution. Further, on a best estimate basis, the small size of the measured boraflex gaps and scattered radial and axial distribution of the measured boraflex gaps is estimated to have had a negligible impact to the K-effective of the Millstone Unit No. 2 spent fuel pool to date.

1c. Boraflex degradation is determined as follows:

First, the EPRI RACKLIFE computer program is used to predict the level of degradation in each of the boraflex panels in the Millstone Unit No. 2 spent fuel pool. The RACKLIFE computer program calculates the integrated gamma dose to each boraflex panel by tracking individual fuel assembly residence time and burnup history in each storage location. Knowing the gamma dose to each boraflex panel, the spent fuel pool silica measurements, cleanup system parameters, various other water chemistry parameters and rack and pool characteristics, RACKLIFE calculates the projected amount of degradation to each boraflex panel. The RACKLIFE model is updated at regular intervals. The attached Figure 1 shows the Millstone Unit No. 2 spent fuel pool silica measurements over the last decade, with each diamond on Figure 1 representing 1 silica measurement. As can be seen from Figure 1, the measurements have generally been between 1-3 PPM silica over the last 10 years. The Millstone Unit No. 2 racks have been in service for about 15 years, and silica levels this low imply minimal boraflex degradation. The RACKLIFE model results show the average boraflex panel is reduced in Boron-10 density by a value of about 0.8%, with a peak value in the worst boraflex panel of about 0.9%. These are very low predicted boraflex degradation levels, consistent with (and driven by) the observed silica measurements.

Next, based on the RACKLIFE model results, it is determined which boraflex poison panels have the highest predicted gamma dose and boraflex degradation values. Based on this determination, a boraflex poison box is selected for

removal. This in-service boraflex poison box is then removed from the Region A or B racks, and then the boraflex is examined and destructively tested. This is accomplished by cutting away the stainless steel protective layer, examining the boraflex, selecting boraflex sections for removal, and then sending the selected boraflex sections offsite for testing. Both a visual examination and material testing is performed on the boraflex. Material testing includes neutron attenuation testing at selected locations of this in-service boraflex for any loss of B-10 density.

Most spent fuel rack designs do not have this Millstone Unit No. 2 capability to remove and destructively test in-service boraflex panels. Millstone considers this preferable to BADGER testing. BADGER testing performs neutron attenuation testing in an in-situ environment. At Millstone Unit No. 2, BADGER testing would have to deal with highly borated spent fuel pool water and the Millstone 2 flux-trap design. These issues make test results from BADGER less accurate. In contrast, removal of in-service boraflex material allows direct visual observation of how actual in-service boraflex material is performing. Further the boraflex material is removed and tested off-site for detailed material condition, including B-10 density, under more controlled and accurate conditions. Thus while BADGER testing provides only measured B-10 areal density as its result, our destructive examination method provides measured B-10 areal density as a result, as well as the ability to visually observe what is happening to the boraflex material. The advantage of BADGER testing over our method of destructive boraflex examination, is that BADGER testing can test more boraflex panels. However, given the low level of boraflex degradation described above, we consider our current method of removal and destructive testing of in-service boraflex material to be better than BADGER testing in ensuring that the boraflex material is performing consistent with criticality analysis credit for boraflex. Should boraflex degradation levels become more appreciable, it is possible that a switch from our current method to BADGER testing may be appropriate to increase the number of boraflex panels tested. That determination, if necessary, would be made as a result of our continuing boraflex monitoring program.

- 1d. The RACKLIFE predictive model results show average boraflex panel degradation values of about 0.8% Boron-10 areal density loss, and peak boraflex panel degradation values of about 0.9% Boron-10 areal density loss.

The most recent results (year 2000) of destructive examination of in-service boraflex material has shown no significant boraflex degradation. The boraflex removed was in the spent fuel pool since the racks were installed, which corresponds to about 15 years in the Millstone Unit No. 2 spent fuel pool. The specific panel chosen was one that RACKLIFE determined to have one of the highest gamma doses, and one of the highest B-10 areal density losses. From this full length boraflex panel, 4 sections of near full boraflex width, and 12" to 15" in boraflex length were removed from the boraflex panel for offsite B-10 density

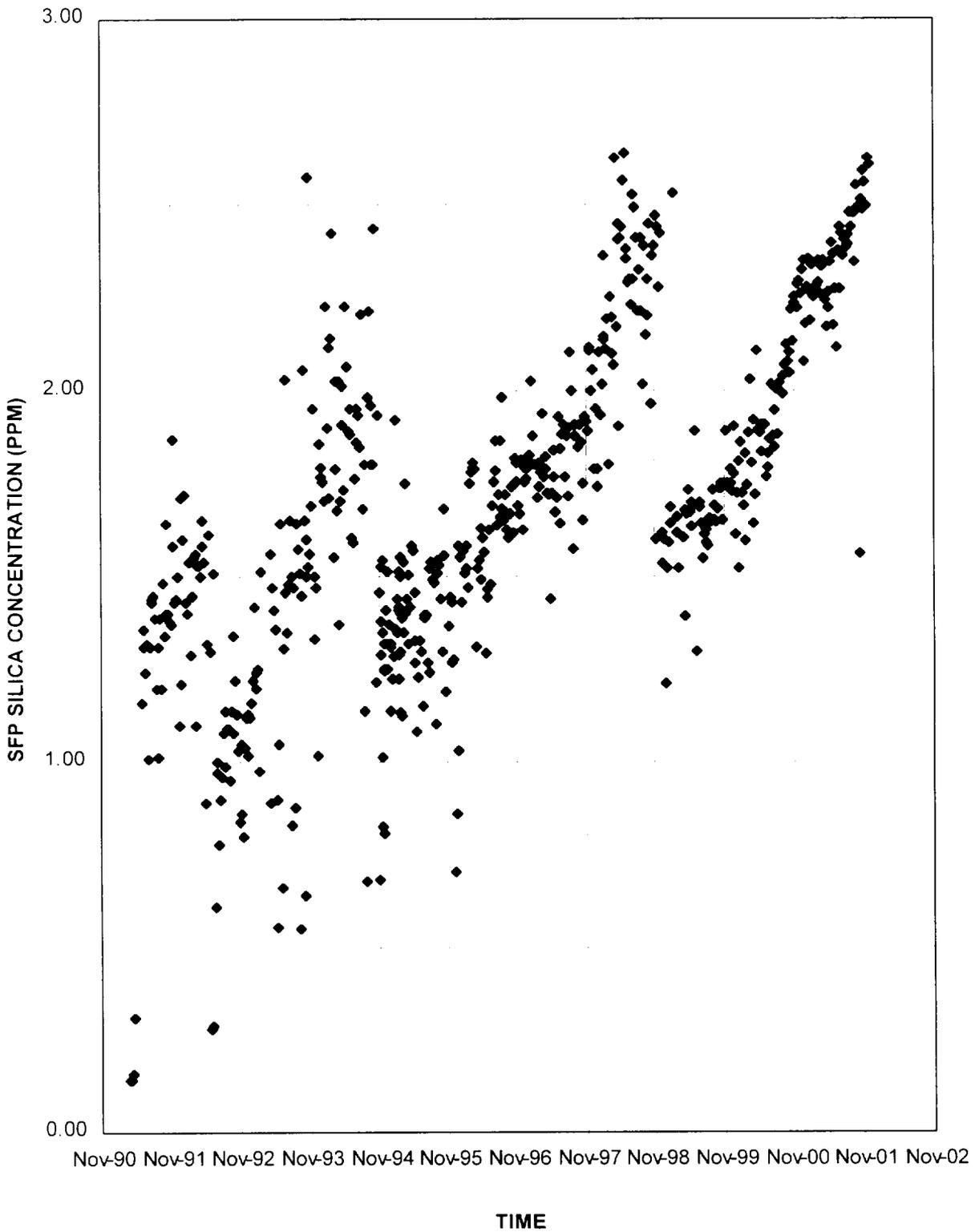
measurements. The 4 boraflex sections chosen comprised about 1/3 of the entire boraflex panel. The 4 sections were chosen to avoid the top and bottom 2 feet of the boraflex, which would have low gamma dose. Also sections of boraflex were chosen if they showed thinned areas. Some minor thinning was observed in small areas, and the worst of these minor thinned areas was included for later B-10 density testing. Boron-10 density testing (neutron attenuation testing) was performed by Northeast Technology Corporation on the 4 boraflex sections. This testing was performed by selecting 32 random locations, and 1 specific location for a section that showed some minor thinning. The lowest Boron-10 areal density of any of the 33 measurements was found to be 0.03532 grams Boron-10/cm².

The Millstone Unit No. 2 boraflex B-10 areal density was manufactured to be a nominal 0.033 grams Boron-10/cm², +/- 0.003 grams Boron-10/cm². Thus, the measured boraflex material B-10 areal density is still within the manufacturer's range of expected values. The loss of 0.8-0.9% B-10 density (0.033*0.009 = 0.0003 grams Boron-10/cm²) predicted by RACKLIFE would be too small to see due to the manufacturing tolerance variation. Also, because of boraflex shrinkage, it is possible for measured (actual) B-10 areal density to increase slightly.

- 1e. The Millstone Unit No. 2 boraflex B-10 areal density was manufactured to be a nominal 0.033 grams Boron-10/cm², +/- 0.003 grams Boron-10/cm². The criticality analysis that supports this license amendment credits a minimum boraflex B-10 areal density of 0.025 grams Boron-10/cm². This corresponds to about a 25% reduction from the nominal B-10 areal density of 0.033 grams Boron-10/cm². This reduced B-10 areal density value of 0.025 grams Boron-10/cm² was used to provide future margin, should unexpected reductions occur in boraflex B-10 density. As described above in the answer to question 1d, the current measured boraflex B-10 density is still within the manufacturing tolerance.

Boron-10 density measurements from the periodic removal and testing of in-service boraflex poison boxes will be used to verify compliance with the criticality analysis assumption of a minimum boraflex B-10 areal density of 0.025 grams Boron-10/cm². RACKLIFE predicted values of Boron-10 density are used as an independent check of the B-10 measurements from the periodic in-service boraflex poison box removal.

FIGURE 1 MILLSTONE 2 SFP SILICA VS. TIME



Question 2

Your submittal stated that in 1992, the NRC approved an amendment that allowed 40 cells in Region B to be blocked for the purpose of compensating for an error in a previous criticality analysis. The licensee now proposes to store spent fuel underneath the cell blockers. Placing spent fuel underneath the cell blockers results in a configuration not originally intended for the spent fuel pool. The staff requests the licensee to describe the program that verifies the material condition of these cell blockers and the cells that have been blocked for approximately 10 years.

Response

The proposed Technical Specification (TS) Limiting Condition For Operation (LCO) 3/4.9.19 ensures that cell blockers are in place, as necessary, to ensure compliance with the criticality analysis. Beyond this LCO, there is no need for a program for verifying the material condition of cell blockers, or for verifying the material condition of the fuel storage cells under the cell blockers. The reasons for this are described below.

Cell Blockers:

Cell blockers serve no function other than to provide a visible cue to the fuel handler that fuel should not be inserted in that location. As long as the cell blocker is in place and can be seen, it is serving its design function. The cell blocker has no structural credited feature. A cell blocker is less than 11 inches long and fits mostly inside the storage cell, resting on the top of the poison box. For reference, a fuel assembly sits inside a poison box, and the poison box sits inside the storage cell. The top of the cell blocker sits a few inches above the rack to enhance visibility. The cell blockers are of an open design, with more than $\frac{1}{2}$ of the cross sectional area at the top of the blocker open to flow. Thus heat removal is not inhibited for that storage location. When we use the term "cell blocker," that does not mean the fuel storage cell is isolated in any respect other than the inability to insert a fuel assembly in that location. Spent fuel pool water can freely flow through the vacant cell. The cell blocker is made of stainless steel and weighs about 20 pounds. The cell blockers were designed to be removable, and were designed to allow fuel to be stored underneath them. The existing TS LCO (TS 3/4.9.19) has been carried forward to require cell blockers be in place, or if not in place, the LCO requires certain fuel storage cells to be empty. Since the cell blocker is made of all stainless steel, there are no material concerns in the spent fuel pool environment. Since the cell blocker sits inside the rack location on top of the poison box, there are no mechanical connections to inspect. Since the blocker has no design function other than to be visible, which is effectively met by TS LCO 3/4.9.19, there is no additional testing necessary. Thus we do not see that any inspection of the cell blockers is necessary to carry out the proposed changes.

Storage cells:

As noted in the question, it is true that for Region B, the original rack design did not contemplate cell blockers above stored fuel. As described in our submittal dated November 6, 2001, (Attachment 1 page 12, "Increased Fuel Storage"), all region B storage cells were originally designed to store either intact fuel assemblies, or consolidated fuel storage boxes. The ability of a fuel storage cell to carry out its design function is not however affected by whether a cell blocker is or is not in place. A cell blocker is not mechanically attached to a storage cell. The approximate 20 pound weight of a cell blocker in every 4th location is negligible compared to the fact that each fuel storage location in Region B was designed to carry a consolidated fuel storage box. However, consolidated fuel storage boxes are only allowed by TS to be stored in Region C. Thus for Region B, the weight of a fuel assembly, plus insert (control rod, etc), plus blocking device, is well below the weight of a consolidated fuel storage box, thus the Region B design is very conservative. With regard to heat removal, as we discussed in the submittal, the cell blockers are of an open design, such that heat removal is not inhibited, whether or not there are fuel assemblies stored underneath the cell blockers. When we use the term "cell blocker" that does not mean the fuel storage cell is isolated in any respect other than the inability to insert a fuel assembly in that location. Spent fuel pool water can freely flow through the vacant cell. Thus, storage of fuel under the cell blockers does not invalidate or degrade the ability of the Region B fuel storage racks to safely store fuel.

With regard to whether a material inspection is necessary for those fuel storage locations which have not been used for 10 years, there is no difference between: (1) the insertion of fuel into a storage location which had a cell blocker and was vacant for 10 years, or (2) insertion of a fuel assembly into a fuel storage cell which never had a cell blocker, and has been vacant for 10 years, solely because the storage location was not yet needed. The presence of a cell blocker has not led to the storage location being isolated in any way, other than that a fuel assembly could not be inserted. The blocked fuel storage location is subject to the same spent fuel pool water conditions as adjacent fuel storage locations. Spent fuel pool water can freely flow through the inlet holes at the bottom of the rack cell, and out the top where the cell blocker has more than ½ of its cross sectional area open to flow, allowing the "blocked" fuel storage location to be in good contact with the pool environment. Thus in summary, the material condition of the blocked storage rack cells should not be different than the condition of the unblocked storage rack cells.

As additional information, this license amendment request is similar to TS amendment 172 in 1994, in which Millstone Unit No. 2 was authorized to remove the 234 cell blockers in Region C, and place fuel in these 234 previously blocked locations. In this case the cell blockers could be permanently removed. These cell blockers were in place for about 9 years. These Region C cell blockers were of a slightly different design, but similar in the respect that the blocker was of an open design such that heat

removal would not be restricted. These 234 fuel storage locations were also originally designed to store either intact fuel assemblies, or consolidated fuel storage boxes. No inspection of the fuel storage cells was performed or necessary, since the vacant storage cell was unaffected by the presence of the cell blocker. No problems were encountered in inserting these 234 fuel assemblies into the previously blocked locations, or any other problems since then, related to removal of the cell blockers.

Question 3

Your submittal stated that the changes to current Technical Specification (CTS) 3/4.9.18 make the specification consistent with NUREG 1432 "Standard Technical Specifications - Combustion Engineering Plants" (STS) 3.7.18. The changes made to CTS 3/4.9.16.2 and 3.9.17 to proposed Technical Specification (PTS) 3/4.9.17 make the PTS similar to STS 3.7.17 adjusted for plant specific nomenclature and design. However, the PTS differs from the STS in that the applicability does not have the STS words "and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool" and Required Action A.2.2 is not present. Since the changes to these specifications seemed to be based on the STS with the intent on consistency, no justification is provided for this deviation from the STS. Provide a discussion and justification for this deviation from STS 3.7.17.

Response

The concern of this question, is that the proposed TS 3/4.9.17 did not contain the wording associated with "and a fuel storage pool verification ..." (and associated required action A.2.2), that is contained within STS 3.7.17, and this deviation should be explained.

The proposed TS 3/4.9.17 is more conservative than STS 3.7.17. The proposed TS 3/4.9.17 requires the boron concentration to be maintained at ≥ 1720 PPM whenever fuel (fuel assembly or consolidated fuel storage box) is stored in the spent fuel pool. In contrast, the boron concentration requirements of STS 3.7.17 is applicable only and limited to "When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool." Thus if a fuel storage pool verification has been performed, there is no requirement to maintain any boron in the spent fuel pool per STS 3.7.17 and the TS will not be applicable. As explained in the STS 3.7.17 bases, this is because STS 3.7.17 has as the underlying basis that there is the potential for a dropped or misplaced fuel assembly. Thus by STS 3.7.17 logic, if no fuel is being moved and a fuel pool verification has been performed, there is no further need for a spent fuel pool boron concentration requirement. The proposed TS 3/4.9.17 is more conservative than STS, and requires the spent fuel pool boron concentration to be maintained "Whenever any fuel assembly or consolidated fuel storage box is stored in the spent fuel pool." Further, as described in the submittal, 1400 PPM of soluble boron is sufficient to ensure

that all fuel handling/misplacement events or heavy load accidents do not allow K-effective to reach 0.95. The proposed TS 3/4.9.17 requires ≥ 1720 PPM of soluble boron (more than the 1400 PPM needed), "Whenever any fuel assembly or consolidated fuel storage box is stored in the spent fuel pool". Therefore, this conservatively ensures that sufficient soluble boron concentration is present should a fuel handling/misplacement event or heavy load accident occur. In summary, while we tried to be as consistent with STS as possible, the proposed TS 3/4.9.17 is more conservative.

Question 4

CTS 4.9.16.2 and 4.9.17 specify that the spent fuel pool (SFP) boron concentration be verified to be within the specified limit within 24 hours prior to movement of a fuel assembly, a consolidated fuel storage box or a shielded cask. In addition, CTS 4.9.17 also requires that the boron concentration be verified every 72 hours during the movement of fuel assemblies and consolidated fuel storage boxes. PTS 4.9.17 retains the 24-hour verification and also requires that the boron concentration be verified every 7 days. The justification for the 7-day frequency states that it is a more conservative requirement than the current requirements because the verification is performed whenever fuel is stored in the pool and there is no corresponding existing surveillance. The staff agrees that this change is a more restrictive change except for that time period when fuel assemblies are stored in the pool and they are being moved. In this case, there is a corresponding existing surveillance whose frequency is every 72 hours. For this aspect of the change, the change is less restrictive (72 hours to 7 days). No justification is provided for this less restrictive change. Provide a discussion and justification for this less restrictive change.

Response

The proposed boron concentration surveillance time interval is 7 days whenever fuel (fuel assembly or consolidated fuel storage box) is stored in the spent fuel pool. This question concerns the justification for the longer time interval for spent fuel pool boron surveillance from 72 hours to 7 days, when fuel is stored in the spent fuel pool and fuel is being moved. The justification for this 7 day surveillance interval was stated in our submittal dated November 6, 2001, Attachment 1, Page 16, first 4 paragraphs. Also the bases of proposed TS 3/4.9.17 (last paragraph) discuss the basis for the 7 day surveillance.

A 7 day surveillance interval is acceptable because no deliberate major replenishment of pool water is expected to take place over this short period of time (7 days). Whether fuel is being moved or not moved in the spent fuel pool has no effect on whether there will be a deliberate major replenishment of pool water. Thus the movement of fuel in the spent fuel pool has no relationship to the underlying basis of the 7 day surveillance interval. Therefore it is acceptable to allow a boron surveillance interval of 7 days vs.

the current 72 hours, when fuel is being moved in the spent fuel pool. This 7 day surveillance is consistent with STS and the bases for STS, as well as Page 15 of the NRC Safety Evaluation Report (SER) (see our submittal dated November 6, 2001, Attachment 1, reference 2) for WCAP-14416-P.

Question 5

CTS 4.9.16.2 and 4.9.17 specify that the SFP boron concentration be verified to be within the specified limit within 24 hours prior to movement of a fuel assembly, a consolidated fuel storage box, or a shielded cask. PTS 4.9.17 retains the 24-hour verification. The submittal provides an analysis for a SFP boron dilution accident. Based on this analysis, one could conclude that boron concentration verification prior to movement should be retained, but that the frequency "within 24 hours prior to movement" should be changed to a frequency that is less than 24 hours. One could also conclude based on the discussion provided in the analysis on SFP water level controls, alarms, and operating procedures and the revised Bases for 3/4.9.17, that this surveillance is not necessary. Provide a discussion and justification on the necessity of this surveillance and the adequacy of the specified frequency.

Response

This question pertains to the need for the surveillance on boron concentration within 24 hours prior to fuel or cask movements.

This surveillance on boron concentration within 24 hours prior to fuel or cask movements is not technically necessary, but was retained only because it was in the existing TS, and as a conservative measure, we determined to retain it. There is no technical basis for the 24 hour surveillance. This 24 hour surveillance is not needed to detect a boron dilution event. Boron dilution event protection is provided by: (1) the 7 day surveillance of boron concentration ensures that 1720 PPM is available as the pre-condition to the dilution event, and (2) as noted in the NRC question, protection for actual detection of the boron dilution event is provided by spent fuel pool water level alarms and operating procedures. Retention of this 24 hour surveillance is recommended only as a conservative measure.

Question 6

Your submittal stated that the changes to CTS 3/4.9.18 make the specification consistent with STS 3.7.18. PTS 3.9.18(a), (b)(1), b(2), and (c) do not seem to be consistent with STS 3.7.18. STS 3.7.18 contains the words "or in accordance with specification 4.3.1.1." The changes made in converting CTS 5.6.1 to PTS 5.6.1.e), f), g) and h) specify the criteria for storing fuel assemblies in Regions A, B, and C of the SFP similar to the storage criteria specified in STS 4.3.1.1.e and f. No justification for this deviation to the STS is provided. Revise PTS 3.9.18(a), (b)(1), (b)(2) and (c) to be

consistent with STS 3.7.18, as modified by plant specific characteristics, or provide a discussion and justification for this deviation.

Response

The first part of this question is to explain why the words "or in accordance with Specification 4.3.1.1", from STS 3.7.18, were not used in proposed TS 3.9.18. The reason for this is that STS 3.7.18 does not require these words if the Figure number is used. The bracketed words "or in accordance with Specification 4.3.1.1" denotes that either the Figure number be used, "or" instead of the Figure number, "in accordance with specification 4.3.1.1." We used the Figure number rather than a reference to the Design Function TS section.

The second part of this question concerned proposed TS 5.6.1.e), f), g) and h), which were not completely consistent with STS 4.3.1.1.e and f.

Our submittal did not state that proposed TS 5.6.1 was meant to comply with STS. We stated in our submittal dated November 6, 2001, (attachment 1 Page 6, 3rd paragraph) that the proposed TS 5.6.1 was intended to comply with the NRC SER (our submittal, Attachment 1, reference 2) contained in WCAP-14416-NP-A. This NRC SER states that "Licensees using the Westinghouse Improved STS... should adopt specification 3.7.16... and 4.3.1... as shown in section 5.0 below." The Section 4.3.1 listed in this SER refers to the Design Features TS section. While we do not use Westinghouse Improved STS, we tried to comply with the wording as close as was possible, since this wording was specifically designed for use with soluble boron credit. However, since the design features section 4.3.1(f) and (g) listed in this NRC SER also has similar wording to the STS 4.3.1.1.e and .f, the underlying NRC question is still valid.

The proposed Millstone Unit No. 2 design features section 5.6 could not be made consistent with the design features section 4.3.1(f) and (g) listed in this NRC SER, because we have multiple burnup vs. enrichment curves in the same region, storage of fuel under cell blockers in Region B, and consolidated fuel. Therefore, for proposed TS 5.6.1.e), f), g) and h), which correspond to section 4.3.1(f) and (g) listed in this NRC SER, we maintained as similar wording as possible to the current TS 5.6.1.b), c), d) and e). The information contained in proposed TS 5.6.1.e), f), g) and h) provides the same information contained in NRC SER section 4.3.1(f) and (g), but is stated in wording similar to existing TS wording to account for the plant specific differences.

Question 7

Your submittal stated that the changes to CTS 3/4.9.18 make the specification consistent with STS 3.7.18. PTS 3.9.18(a), (b)(1), (b)(2), and (c) do not seem to be consistent with each other and with the STS. PTS 3.9.18(a) and STS 3.7.18 use the same words "The combination of initial enrichment and burnup of each fuel assembly..."

PTS 3.9-18(c) also uses the same words except "each fuel assembly" is changed to "each consolidated fuel storage box". PTS 3.9.18(b)(1) and (b)(2) substitute the word "a" for "each" in the phrase "each fuel assembly". No justification is provided as to why this substitution was made in PTS 3.9.18(b)(1) and (b)(2). The change could limit the number of fuel assemblies with an initial enrichment and burnup within the specified limits to be stored in Region C to one fuel assembly. Revise PTS 3.9.18(b)(1) and (b)(2) to be consistent with STS 3.7.18, or provide a discussion and justification for this deviation.

Response

First, this question asks about PTS 3.9.18(c), which uses the words "each consolidated fuel storage box", instead of "each fuel assembly" as in PTS 3.9.18(a) and STS 3.7.18.

In answer to this question, the STS do not address consolidated fuel storage boxes (CFSBs), however, the Millstone Unit No. 2 TS requirements for CFSBs must be addressed. A description of a CFSB and the current pool configuration was provided in our submittal dated November 6, 2001, Attachment 1 page 2 under the paragraph labeled "Current Millstone 2 Spent Fuel Pool Configuration". Since the CFSBs have their own limits (TS Figure 3.9-3) they must be listed separately from other fuel assemblies. It is noteworthy that the TS limits for the CFSB have not been changed by the proposed changes. CFSBs are only allowed to be stored in Region C and TS Figure 3.9-3 has not been changed. Proposed TS 3.9.18(c) therefore must use the words "each consolidated fuel storage box" so that the separate requirements of CFSBs are listed.

A second question which is asked is why does PTS 3.9.18b(1) and 3.9.18b(2) use the words "a fuel assembly" instead of "each fuel assembly"? A further question is raised concerning the possibility that this may limit the fuel storage of Region C to one fuel assembly.

In answer to this second question, PTS 3.9.18b(1) and 3.9.18b(2) use the words "a fuel assembly" instead of "each fuel assembly" because a fuel assembly can be stored in Region C either with or without poison pins. Thus there are some fuel assemblies stored in Region C that have poison pins, and some fuel assemblies that are stored in Region C without poison pins. If the word "each" was used instead of "a", then PTS 3.9.18b(1) and 3.9.18b(2) would literally require that "each" fuel assembly must meet PTS 3.9.18b(1), or "each" fuel assembly must meet PTS 3.9.18b(2). This would mean that all fuel assemblies stored in Region C must contain poison pins, or all fuel assemblies stored in Region C must not contain poison pins. This would preclude allowing a mix of fuel assemblies with and without poison pins to be stored in Region C. The existing TS 3.9.18 clearly allows a mix of fuel assemblies with and without poison pins to be stored in Region C. The use of the word "a" instead of "each" for fuel assembly stored in Region C avoids this problem. The use of the word "a" requires that

an individual fuel assembly must meet either PTS 3.9.18b(1) or 3.9.18b(2). Millstone does not interpret that the use of the word "a" limits the number of fuel assemblies allowed to be stored in Region C to one fuel assembly. The proposed wording simply requires that "a" fuel assembly stored in Region C must meet either PTS 3.9.18b(1) or 3.9.18b(2). Also, Proposed TS 5.6.1(g) and 5.6.3 clearly acknowledge that more than one fuel assembly may be stored in Region C.

Question 8

CTS 5.6.1 is replaced by PTS 5.6.1 which is patterned after STS 4.3.1 as modified by plant specific nomenclature and design characteristics. PTS 5.6.1.c) and d) specify the K_{eff} s for the SFP if flooded with unborated and borated water respectively. The K_{eff} s include an allowance for uncertainties as specified in PTS 5.6.1.c) and d). However, no mention or reference to a document is made as to where these uncertainties can be found as is done in STS 4.3.1. Revise PTS 5.6.1.c) and d) to reference the document(s) where the allowance for uncertainties can be found, or provide a discussion and justification for this deviation.

Response

This question concerned why there was no specific reference in proposed TS 5.6.1.c) and d) for approved uncertainties, while STS 4.3.1 has specific wording for referencing where the approved uncertainties are.

Our submittal dated November 6, 2001, did not state that proposed TS 5.6.1 was meant to comply with STS. We stated (Attachment 1 Page 6, 3rd paragraph) that the proposed TS 5.6.1 was intended to comply with the NRC SER (Attachment 1 reference 2) contained in WCAP-14416-NP-A. This NRC SER states that "Licensees using the Westinghouse Improved STS... should adopt specification 3.7.16... and 4.3.1... as shown in section 5.0 below." The Section 4.3.1 listed in this SER is similar to STS 4.3.1, but not exactly the same. While we do not use Westinghouse Improved STS, we tried to comply with the wording as close as was possible, since this wording was specifically designed for use with soluble boron credit. However, since 4.3.1.1.b and c. listed in this NRC SER also has the wording to list where the approved uncertainties are located, the NRC's basic question is still valid.

We stated in our submittal dated November 6, 2001, Attachment 1 page 8 paragraph 3, that we could not put references for uncertainties in proposed TS 5.6.1 because of the NRC's requirement stated in NRC letter dated July 27, 2001, (reference 3 of our submittal). This NRC letter stated:

"However, as a result of identified non-conservatism in a Westinghouse Topical Report (TR) on this subject, future licensing submittals from licensees will no longer be able to reference the methodology in the affected document.", also:

"Therefore, the staff concludes that the methodology approved in WCAP-14416 can no longer be relied upon as "approved methodology" by the NRC staff ... For Future licensing actions, licensees will need to submit plant specific criticality calculations for spent fuel pool configurations that include technically supported margins."

Thus we could not add previously approved references to proposed TS 5.6.1.c) and d), because there are none. Our submittal, Attachment 1 page 8 paragraph 3 states this. The uncertainties used in this analysis were contained in our submittal, Attachment 5, which contains the Westinghouse criticality analysis. The NRC approval for uncertainties will be via NRC approval of this licensing amendment. Therefore, we did not see how we could reference a license amendment in advance of issuance, and, as a result we added no specific approved reference to TS 5.6.1.c) and d).

Question 9

On page 6 of Attachment 1 of your submittal, the paragraph related to Design Feature 5.6.1a stated that the proposed change "acknowledges an NRC requirement" stated in an SER related to Westinghouse topical report WCAP-1 4416-P. In addition, the paragraph related to Design Features 5.6.1 b through 5.6.1 h stated that "[t]he wording format of these proposed design features is intended to comply with the NRC SER contained in WCAP-14416-NP-A." Please state the requirements and the design features eluded to in the SER.

Response

With regard to proposed Design Feature TS 5.6.1.a, we stated the proposed change "acknowledges an NRC requirement" in the NRC's SER (reference 2 of our submittal dated November 6, 2001) for WCAP-14416-P. This refers to the limit in proposed TS 5.6.1.a for maximum fuel pin enrichment. The NRC's SER (Page 11) for WCAP-14416-P states: "(2) The maximum fuel rod enrichment shall be limited to 5.0 w/o U-235." We felt that given this SER restriction, and the fact that Millstone Unit No. 2 fuel could (and does) have multiple fuel pin enrichments, that we should include TS limits on both fuel assembly average enrichment and maximum individual fuel pin enrichment.

Also, concerning proposed Design Features TS 5.6.1b through 5.6.1h, we stated that "the wording format of these proposed design features is intended to comply with the NRC SER contained in WCAP-14416-NP-A". This NRC SER states (page 12, 3rd paragraph) that "Licensees using the Westinghouse Improved STS... should adopt specification 3.7.16... and 4.3.1... as shown in section 5.0 below." Section 4.3.1 is the design features TS. Thus, while we do not use Westinghouse Improved STS, (Millstone Unit No. 2 is a CE NSSS) we are complying with the wording of section 4.3.1 of this NRC SER as close as was possible in the proposed TS 5.6.1b through 5.6.1h, since it was specifically intended to be used for plants adopting soluble boron credit.

When it was not possible to use the same wording as the NRC SER section 4.3.1, due to plant specific differences, we retained existing TS wording as close as was possible. As a result, proposed TS 5.6.1b), c) and d) were written in conformity to the NRC SER, except as noted in the response to questions 8 and 9. Proposed TS 5.6.1a), e), f), g) and h) were written conforming to the existing TS wording, due to the significant plant specific differences relative to the NRC SER section 4.3.1.

Question 10

In the last paragraph on page 8 of Attachment 1 of your submittal, it is stated that individual pins may have an enrichment as high as 5.0 w/o U-235. Does every assembly containing these highly enriched individual pins have an assembly average that is more conservative than accounting for individual pins?

Response

Many (but not all) Millstone Unit No. 2 fuel assemblies utilize multiple fuel pin enrichments in the radial direction. This is necessary primarily to control power peaking due to the large waterholes present in vacant Control Element Assembly (CEA) guide tubes. As a result, lower enrichment fuel pins are placed around the CEA guide tube locations, and higher enrichment fuel rods are placed elsewhere. Also some reduced enrichment fuel pins are necessary in fuel rods containing gadolinia. Thus while some individual fuel pins could have enrichments as high as 5.0 w/o U-235, other fuel pins must have lower enrichments to comply with the requirement that the radial average enrichment not exceed 4.85 w/o U-235. The use of a radial average enrichment in the criticality analysis is conservative relative to modeling the individual fuel pin enrichments that make up the average. The reason for this is the fuel pins surrounding the water holes have a higher neutron importance. Thus for criticality calculations, use of a radial average enrichment for all fuel pins (instead of the individual pin enrichments) results in higher fuel pin enrichments around the water holes, maximizing the calculated K-effective value. Thus, for all current Millstone Unit No. 2 fuel designs which utilize radial fuel pin enrichment zoning, it is conservative to use the average fuel pin enrichment.

A similar discussion of the above was contained in the Westinghouse criticality analysis contained in our submittal, Attachment 5, Page 47, section 3.5.

Question 11

On page 9 of Attachment 1 of your submittal, in the middle of the fourth paragraph, it is stated that the boraflex gap model has been made significantly more conservative. What exactly was done to accomplish this?

Response

The question is to explain how we made the boraflex gap model significantly more conservative.

Existing Criticality Analysis:

The existing criticality analysis conservatively assumes that 5.65" gaps exist in all boraflex panels. These gaps are assumed to be randomly distributed in the axial direction. Boraflex gap measurements at Millstone Unit No. 2 show these to be very conservative assumptions. As described in the submittal dated November 6, 2001, (Attachment 1 pg. 17), from the last blackness testing campaign at Millstone Unit No. 2, not every boraflex panel had gaps, the maximum gap (2.8") was far less than 5.65", and the gaps appeared to be distributed in a generally random fashion in the axial direction.

Proposed Criticality Analysis:

The criticality analysis supporting this license amendment takes a more conservative approach in the handling of the boraflex gap model. This was done to ensure that should unexpected changes in how the boraflex gaps are axially distributed occur, that the criticality analysis will remain bounding. The submitted criticality conservatively penalizes that 5.65" gaps exist in all boraflex panels. This is the same as the existing analysis. The difference in the proposed boraflex gap model vs. the existing boraflex gap model, is how the gaps are distributed in the axial direction. As stated in the submitted criticality analysis in our letter dated November 6, 2001, (Attachment 5, page 6, section 1.1 (item 2)):

"All Boraflex panels are assumed to contain 5.65 inch gaps. The gaps surrounding individual assemblies are coplanar. The gaps are modeled along the midplane of the active fuel height and are "staggered" in adjacent locations by four (4) inches."

The purpose of this change is to allow the possibility that future blackness testing could show a different and unexpected axial distribution of boraflex gaps. Since it is difficult to predict an unexpected condition, the choice of aligning the boraflex gaps into 2 planes near the core midplane with a 4 inch separation is judgmental. The proposed boraflex gap model does result in significantly higher (and more conservative) calculated K-effective values for a given set of conditions. This is equivalent to saying that less credit is being taken for boraflex due to the use of this proposed boraflex gap model. The proposed model of axial gap distribution is very conservative since it is difficult to foresee how the boraflex gaps could cluster at the same elevations. Hence, the boraflex gap model has been made significantly more conservative.

Question 12

On page 10 of Attachment 1 of your submittal, the last sentence of the third paragraph, states that “[i]f axial blankets are present, then the center zone average enrichment would be used.” Please clarify.

Response

This question requests an explanation of the sentence “If axial blankets are present, then the center zone average enrichment would be used.”

Proposed TS Figures 3.9-1A, 3.9-1B, and 3.9-4, use fuel assembly initial enrichment on the “x” axis for determining the allowable fuel burnup. In cases where a fuel assembly has axial fuel zoning, such as axial fuel blankets, there are 3 axial fuel zones. The center zone contains the highest enrichment fuel, and the end axial zones contain the blanket (lower) enrichment. In this circumstance, the appropriate fuel assembly initial enrichment to be used is the center zone enrichment, which is the largest and most conservative enrichment. Therefore for any given fuel assembly, the axial zone with the highest enrichment fuel should be used for the proposed TS figures. In cases where there is both radial fuel pin zoning and axial fuel zoning, the maximum value of center zone average enrichment should be used for the proposed TS figures. This is consistent with the criticality analysis. This is also not a change from current practice, and is controlled procedurally.

Question 13

In the last paragraph on page 12 of Attachment 1 of your submittal, please clarify the apparent contradiction between the statements made in the following two sentences:

“The SFP heat load analysis of record maximizes heat load by having all SFP storage locations filled with fuel at the end of plant life.”

“The Batch B fuel assemblies have a decay time > 16 years, which is a longer decay time than the decay time at any fuel in the heat load analysis of record.”

Response

As stated in our submittal dated November 6, 2001, Attachment 1, pg. 12, the current design basis heat load analysis for the Millstone Unit No. 2 spent fuel pool was revised in accordance with 10 CFR 50.59. In this heat load analysis, spent fuel pool heat load is maximized by assuming all available fuel storage locations in the spent fuel pool are completely full of fuel assemblies. This includes placing fuel in the 40 storage locations in Region B which are currently blocked. This heat load analysis is carried out to the end of 40 year plant life in the year 2015, in order to maximize the heat load. The fuel

storage locations in the pool are assumed to be filled with the most recently discharged fuel from current and previous refueling outages, with the oldest fuel being removed from the spent fuel pool by an undefined means. This conservative scenario is necessary since by end of plant life in 2015, Millstone Unit No. 2 will have generated more spent fuel assemblies than it has spent fuel pool storage capacity. Thus for this end of life scenario to be reached, some fuel would have to be removed from the spent fuel pool. Under this conservative scenario, the oldest fuel remaining in the spent fuel pool would have been discharged in the year 1987. The Batch B fuel assemblies were all discharged by the year 1985 (16 year decay to year 2001). Thus the Batch B fuel assemblies have a decay time which is longer than the decay time of any fuel in the heat load analysis of record.

In summary, the heat load analysis of record already accounts for the 40 currently blocked locations containing fuel, and the decay time of the Batch B fuel assemblies is larger than any other fuel in the heat load analysis of record. This ensures the spent fuel pool heat load analysis of record is not compromised by allowing 40 Batch B fuel assemblies to be placed under the cell blockers.