Hatch Spent Fuel Pool Modification
Amendment 2

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6.0 INSTALLATION

The HDFSS modules are a free-standing, bottom-supported design, resting on support pads placed onto the floor of the fuel storage pool. The installation program will consist of removing the low-density aluminum racks in the pools, placing the new support pads into prescribed positions, and lowering the new modules into position on their respective support pads.

The initial installation will be in the Hatch 2 pool with the pool wet or dry. Special load-tested lifting fixtures, designed with a minimum safety factor of 3, are used to handle the support pads and the storage modules to minimize dropping any materials. The single-failure-proof reactor building crane will be used to remove the old racks and to lower the new equipment into place.

The Hatch 1 pool, which is filled with water and contains spent fuel, will be reracked similarly after the initial installation of modules in Hatch 2 has been completed. Stored fuel may be transferred to the Hatch 2 pool through the transfer canal to empty the Hatch 1 pool, or may be concentrated away from the rerack work locations. The sequence of the rerack work will be such that no heavy equipment will be transferred over stored spent fuel at any time. The installation equipment is designed to allow installation of modules and pads into a water-filled pool. Following the installation, verification of neutron absorbers will be completed.

For the purpose of estimating manhour requirements and man-rem exposures expected during the Unit 1 rerack work, the following work phase breakdown and sequence of events have been assumed:

Phase I — Move fuel from Unit 1 to Unit 2 (424 fuel elements)--This effort will require a two-man team working on the refueling bridge plus miscellaneous engineering and health physics support. Approximately 10 minutes will be required to move each fuel element.

Phase II — Remove seismic restraints from old racks--This work is expected to require a three-man team working above the pool with long-handled tools. It is estimated that this will require approximately 216 manhours for the actual labor. 2

Phase III — Removal of old rack tie down "swing bolts"--It is estimated that this will require a four-man team working above the pool using long-handled tools. It is estimated that this effort will require 472 manhours.

Phase IV — Removal of old racks--This will also be done using a four-man team working above the pool and will require approximately 2 hr/rack for a total of 236 manhours.

Phase V Decontamination of old racks--This work is expected to require one man at a time operating a hydro-lazer or other decontamination equipment approximately 15 feet from the contaminated rack. This is estimated to require 1 hr/ rack.

Phase VI Support pad installation for new racks--This work will require a four-man team working above the pool. There are 29 support pads to set plus an elevation survey of these pads to be done to check levels. Allowing 2 hours per support pad plus 8 hours for the survey results in a total of 264 manhours.

Phase VII Installation of new racks--This work may be done gradually over a period of time since all racks are not required immediately and because of delivery schedule limitations. However, to install all 17 racks, allowing 6 hours of above pool work per rack using a four-man team, results in approximately 408 manhours. 2

Table 6-1 is a breakdown of the man-rem exposures estimated for each phase of the work as described above. Measurements taken over the Unit 1 spent fuel pool during refueling operations have shown that dose rates normally do not exceed 2.0 mrem/hour with a maximum of 3.0 mrem/hour while handling fuel assemblies. The exposure estimates assume all personnel involved in the change-out work to be continuously exposed to the maximum 3 mrem/hour field. Decontamination work on the old racks is assumed to require exposure to a 50 mrem/ hour field. This estimate assumes divers will not be required.

If the Hatch 2 spent fuel pool should become contaminated prior to its modification, this would have essentially no effect on the above estimate which is for Hatch 1 only. Very little exposure is estimated for the Hatch 2 work assuming an uncontaminated pool (i.e., less than five man-rem). If the pool becomes contaminated before reracking, techniques similar to those envisioned for Unit 1 might be required and similar exposures may be received.

TABLE 6-1

EXPOSURE ESTIMATES FOR UNIT I RERACK WORK (MAN-REM)

	PHASE I	PHASE II	PHASE III	PHASE IV	PHASE V	PHASE VI	PHASE VII
Operations	0.78	0.12	0.17	0.35	0.12	0.40	0.62
Construction	-	0.24	0.53	1.1	3.0	0.59	0.90
Health Physics	0.27	0.12	0.17	0.35	0.53	0.20	0.30
Quality Assurance	0.27	0.12	0.17	0.35	0.12	0.20	0.30
Engineering	0.27	0.12	0.17	0.35	0.12	0.20	0.30
Total (each phase)	1.6	0.72	1.2	2.5	3.9	1.6	2.4

TOTAL (all work): 14 Man-rem

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rod enrichment and burnable poison distributions within the bundle. Fuel pin spacers were not included (a conservative exclusion). The nominal bundle dimensions were used for all cases.

The HDFSS includes defective fuel storage spaces attached externally to some of the storage modules. The geometric layout is shown in Figures 2-11 and 2-12. Analyses have demonstrated the HDFSS k_{eff}<0.95 with all defective fuel storage locations occupied with fuel.

The sensitivity of k_∞ analyses to various changing parameters is implied above. More specific relationships are as follows:

a. Bundle Reactivity (percent U^{235}) - Calculations are based on maximum k_∞ thereby obviating enrichment sensitivity considerations.

b. Stainless steel thickness - Neutron absorption by the two layers of stainless steel comprising the storage tube was included in the criticality calculations using the nominal thicknesses of 0.0355 and 0.090 inch for the inner and outer tubes, respectively. The nominal inner tube thickness has been reduced to 0.0300 inch, and Monte Carlo calculations show that the change in k_∞ is within the statistical uncertainty of the calculation (Case 2, Table 7-1).

c. Water density - Figure 7-5 shows the variation of k_∞ with moderator (water) density. Since the cell is under-moderated the optimum k_∞ occurs at 1.0 g/cc.

d. Storage lattice pitch - An analysis was done using a minimum fuel pitch, represented by the storage tubes touching. Material tolerances in the tubes were taken to maximize the k_∞ of the storage lattice. The result of the analysis is given as Case 6 in Table 7-1. A comparison of Cases 2 and 6 in Table 7-1 shows that within the statistical error bounds there is no significant difference between the results. 2

e. The HDFSS and the BWR fuel to be stored in it are designed and fabricated to prevent significant quantities of air or other gas from being entrapped. Thus, no areas of reduced effective moderator density are created. But even if air were trapped, the effect of reduced density on the under-moderated fuel bundles is to reduce the k_{eff} of the system.

7.4 Postulated Accidents

Several fuel assembly drop accidents have been analyzed. The results are summarized in Table 4-3. The handling of heavy objects in the spent fuel pool area is addressed in Section 11.0 of the accident evalution.

A tornado-generated missile model has been used for the Hatch spent fuel pools (refer to the response to Question 130.19 in the Unit 2 FSAR) that could result in impacting the storage module. The angles in the structural grill system associated with the reactor building tornado relief vent openings have been postulated as a secondary missile source resulting from impact of a plank missile. A maximum of three angles

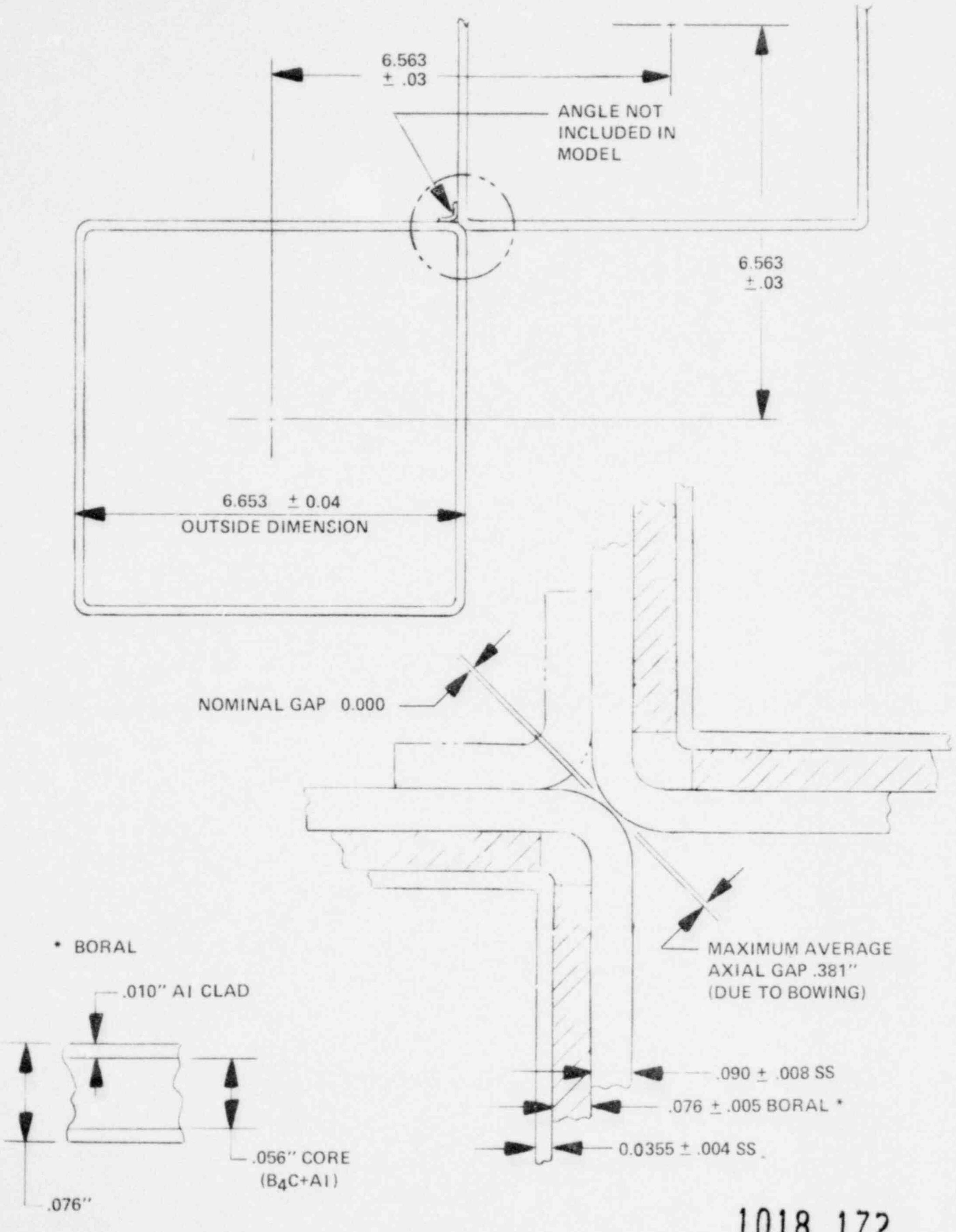

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Figure 7-2 Storage Cell Dimensions

2. Refueling Condition

The pool water is held at 125°F or less with a heat load of 8.5×10^6 Btu/hr generated by stored fuel consisting of a 25 percent core that has decayed for 150 hours since reactor shutdown plus a 25-percent core in storage for one year from a previous refueling operation. The minimum projected time after reactor shutdown to accomplish cooling and opening of the reactor vessel and completion of transferring the spent fuel to the pool is 150 hours. With the assistance of the standby swing cooling train, a combined cooling capacity of 8.5×10^6 Btu/hr is available to cope with the heat generated by newly unloaded fuel and to hold the pool water at or below 125°F.

3. Maximum Condition

The pool water is held to 150°F or less with a heat load of 31.3×10^6 Btu/hr generated by stored fuel consisting of a 100 percent core unloaded from the reactor plus a 25 percent core held over for one year from a previous refueling. The 125 percent core load is assumed to have undergone the following exposures:

25 percent of core:	4 year exposure + 1 year decay
25 percent of core:	4 year exposure + 150 hour decay
25 percent of core:	3 year exposure + 150 hour decay
25 percent of core:	2 year exposure + 150 hour decay
25 percent of core:	1 year exposure + 150 hour decay

2

Under the maximum condition postulated, it is assumed that approximately 150 hours after reactor shutdown the entire core in the reactor will have been transferred to the pool. Thus, the RHR system will be free for cooling the large fuel load in the pool. With the full core offload plus one quarter core remaining from a previous refueling, a single train of the RHR system, without the assistance of SFPC, will maintain the spent fuel pool temperature at or below 150°F 150 hours after the shutdown.

Operating experience with Hatch Unit 1 has indicated that calculated spent fuel pool heat loads and temperatures for the design basis are conservative and the actual heat loads have been approximately 15 percent less than the heat loads calculated.

8.3 Heat Loads and Pool Temperatures for Increased Storage Capacity

8.3.1 To re-evaluate the Plant Hatch spent fuel pool cooling capabilities with the enlarged storage capacity, the decay heat loads were calculated using methods described by Branch Technical Position ASB 9-2 of the Standard Review Plan.

8.3.2 The pool capacity for the increased storage capacity heat load evaluation is assumed to be 5.83 cores. The 5.83 core capacity is arrived

at by assuming 1/4 core yearly offloads to the spent fuel pool up to 5-1/2 cores (22 batches) plus an additional batch (batch 23) of 1/3 core. All batches are assumed to have operated at full power for 90 percent of their four-year exposure time. Also, Reactor Building Closed Cooling Water (RBCCW) system influent temperature to the spent fuel pool heat exchangers is assumed to be 105°F. The three design conditions postulated in Section 8.2 are similarly evaluated below. 2

8.3.2.1 Normal Condition

The heat load analysis for the normal operating condition assumed that there were 22 batches in the pool that had decayed from 1 to 22 years, and the latest batch (23) decayed for 30 days. A single spent fuel pool cooling system train was used for decay heat removal.

The analysis showed that the heat load was 7.24×10^6 Btu/hr and bulk pool water temperature was at or below 139°F. Heat loads and poo temperatures as a function of refueling batches are shown in Figure 8-1.

8.3.2.2 Refueling Condition

The assumptions for the refueling mode analysis were the same as those for the normal mode except that the latest batch was assumed to have decayed for only 150 hours and two spent fuel pool cooling trains were in service.

The analysis showed the heat load was 11.57×10^6 Btu/hr and the bulk water temperature at or below 133°F. Heat loads and pool temperatures as a function of refueling batches are shown in Figure 8-2.

8.3.2.3 Maximum Condition

The analysis for the heat load following full core discharge assumed that the pool already had 19 quarter core batches in storage that had decayed from 1 to 19 years. The calculated heat load from the 19 batches was 2.39×10^6 Btu/hr. The additional decay heat load at 150 hours after shutdown for a full core offload was calculated to be 26.3×10^6 Btu/hr. Therefore, the cumulative heat load in the pool at 150 hours after shutdown is 28.69×10^6 Btu/hr. With a single train of the RHR system aligned for fuel pool cooling duty without the assistance of the SFPC system, the system will maintain pool water temperature at or below 145°F. Figure 8-3 shows the heat load as a function of time after shutdown for the full core discharge.

As an alternative to aligning the RHR system to the spent fuel pool for a full core offload, the fuel may be allowed to decay in the reactor vessel until the heat load of the core has decreased to a point where the SFPC system can maintain a temperature less than the design maximum temperature. A waiting time of 500 hours (approximately 21 days) is required in this case prior to full core offload. After this time, two fuel pool cooling trains can maintain the pool water temperature at or below 150°F (i.e., a heat removal capability of 18.77×10^6 Btu/hr).

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8.3.3 For each design condition analyzed in 8.3.2, completely utilizing the expanded spent fuel pool storage capacity, the present SFPC systems or a single train of the RHR (for the full core offload condition) are capable of maintaining pool water temperatures less than the design maximum temperature of 150°F. Considering the conservative assumptions used in the calculations and past operating experience, the actual temperatures for each condition are expected to be lower than those calculated and described above. A reanalysis of the spent fuel pool heat loads and bulk pool water temperatures was performed for the normal and refueling conditions assuming realistic values of 80 percent plant availability, RBCCW system influent temperature of 95°F to the spent fuel pool heat exchangers, and quarter core discharge batches. The peak bulk pool water temperatures were calculated to be less than 123°F as shown in Figures 8-4 and 8-5. Therefore, there will be no additional releases of radioiodine or tritium from the pool, since the pool temperatures are not raised beyond their original design limit of 125°F. 2

8.4 Loss of Spent Fuel Pool Cooling

The consequences of a loss of the SFPC systems has been evaluated for the following two conditions:

1. Concurrent loss of the SFPC systems.
2. Maximum heat load.

8.4.1 Concurrent Loss of SFPC Systems

Both spent fuel pools are assumed to be loaded as delineated in Section 8.3.2. Unit 1 and Unit 2 are assumed to be shut down for refueling 21 days apart, with Unit 2 being shut down first. Also 21 days is assumed to be the minimum time required to complete a refueling operation. Therefore, Unit 2 will be operating while Unit 1 is shut down. Subsequently, both units' SFPC systems are postulated to be lost 150 hours after Unit 1 is shut down.

Calculations using pool water volumes of 35,640 ft^3 each indicate that the time to boil for the Unit 1 pool is 14.7 hours and that the time to boil for the Unit 2 pool is 22.8 hours. The makeup water requirement following boiling was calculated to be 24 gpm for the Unit 1 pool and 15 gpm for the Unit 2 pool. During transition to boiling, no credit is taken for evaporative heat losses. Water level is maintained by the Seismic Category I Plant Service Water system. Conservatisms are included in the analysis by assuming that all decay heat is rejected to the pool water and none is rejected to the structures. Also, the heat capacity of the makeup water is neglected.

After approximately 150 hours following Unit 1 shutdown, the decay heat contributed by 2/3 core in the Unit 1 reactor pressure vessel has decreased enough to allow aligning one train of the RHR to provide spent fuel pool cooling and reactor pressure vessel cooling. With the reactor vessel head and the spent fuel pool gates removed, the RHR system can be aligned for spent fuel pool and reactor pressure vessel cooling by

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installation of two spectacle flanges and operation of four isolation valves. The time required for realignment is estimated to be 8 hours.

Subsequent to loss of the SFPC systems, Unit 2 will be brought to cold shutdown. A radiological analysis has been performed assuming that both pools boil simultaneously. The consequences are presented in Section 8.6.

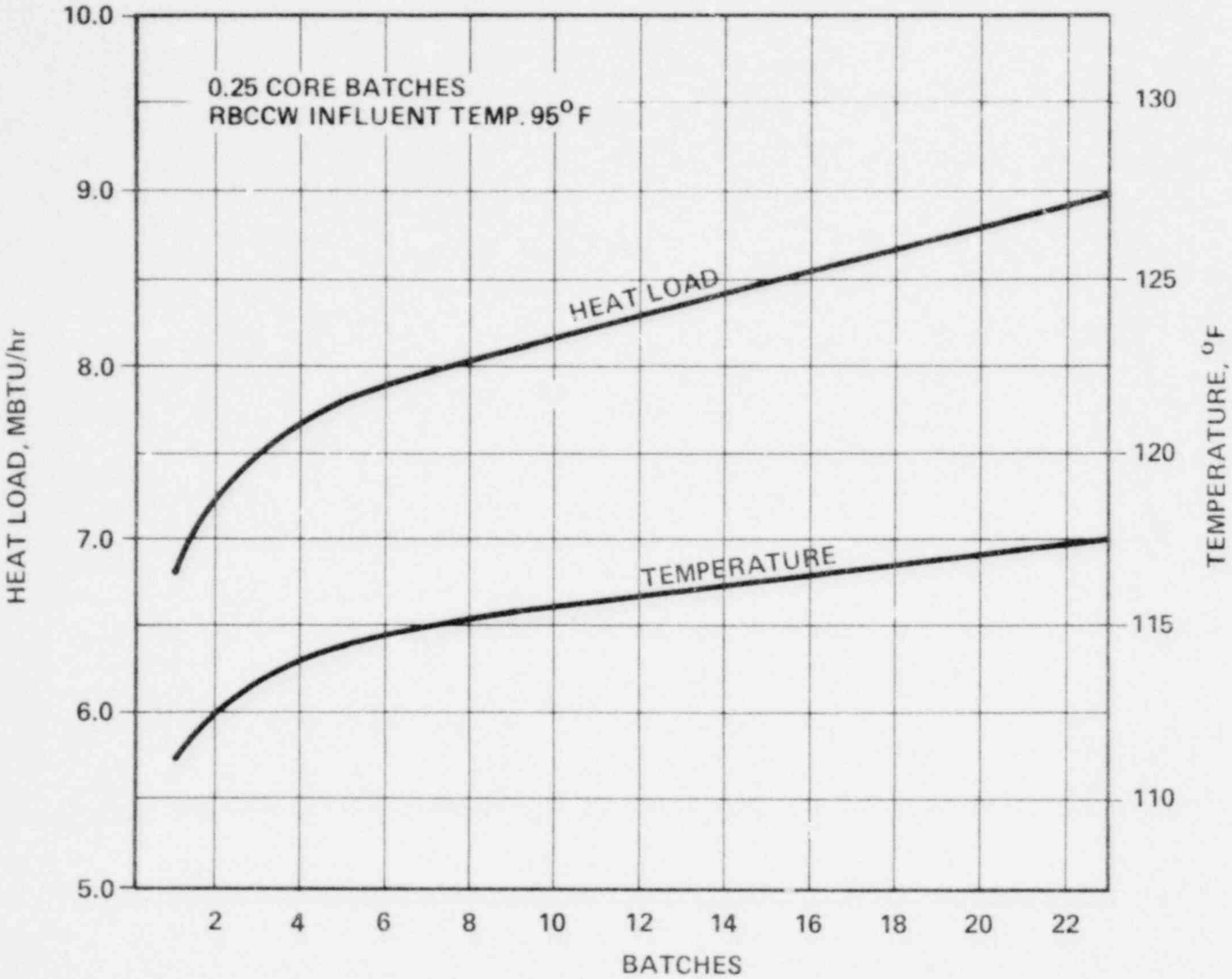

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Figure 8-4 Realistic Fuel Pool Heat Loads and Temperatures 150 hr After Shutdown

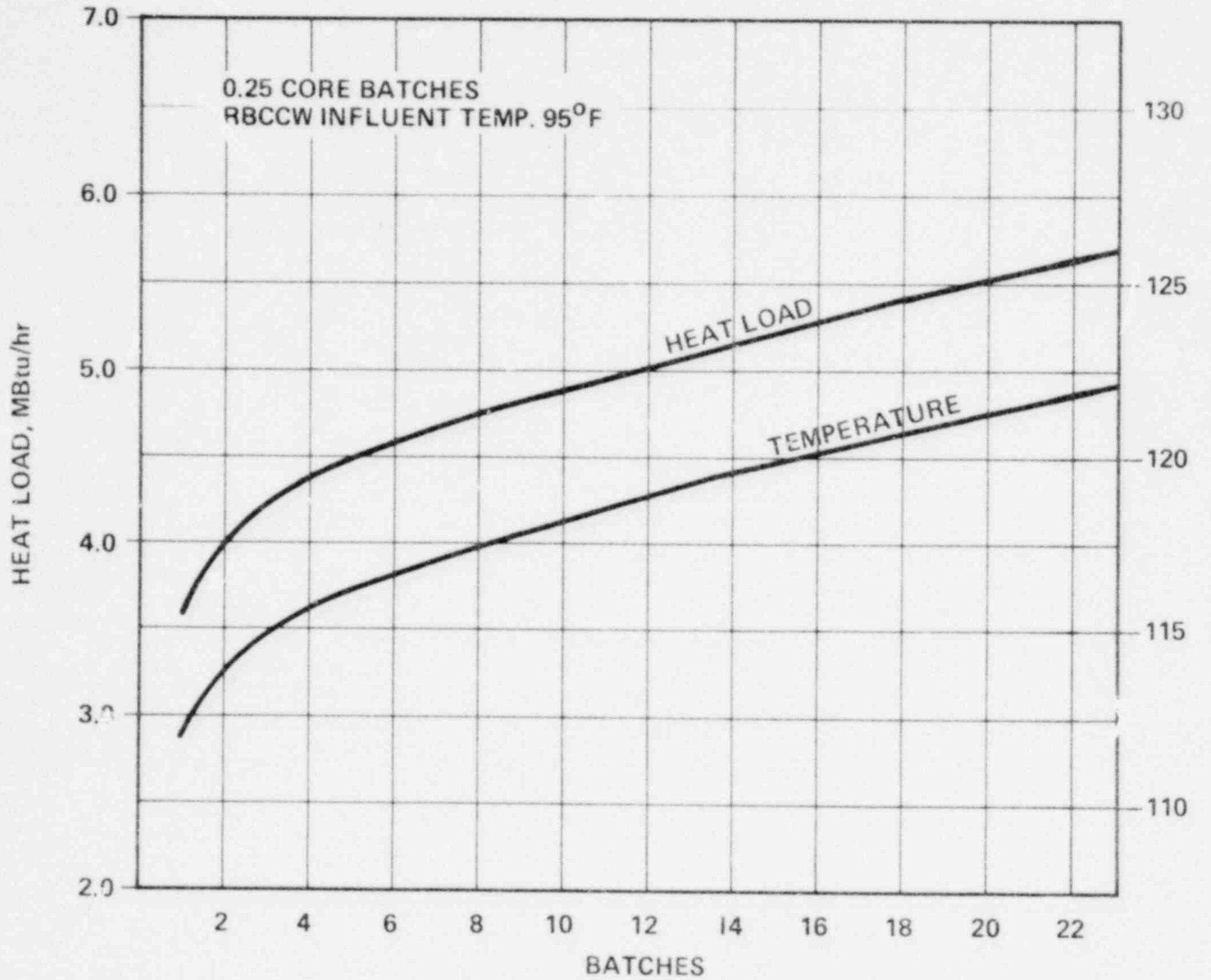

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Figure 8-5 Realistic Fuel Pool Heat Loads and Temperatures 30 Days After Shutdown

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10.0 RADIOLOGICAL EVALUATION

10.1 Spent Resin Waste

The fuel pool filter-demineralizer units are designed to maintain a water conductivity of less than 0.5 micro mho/cm. The units are backwashed when either the differential pressure across the demineralizers is greater than 10 psi or the effluent conductivity is greater than 5 micro mhos.

Hatch Unit 1 experience indicates that the filter-demineralizer was backwashed 41 times during 1978. Each backwash cycle amounts to 2.5 cubic feet of spent resin. The dose attributed to handling of the spent fuel pool resin in the radwaste system is approximately 0.3 man-rem/yr.

The increase in occupational exposure to personnel from the additional fuel assemblies themselves which could be stored as a result of the increased storage capacity resulting from this modification is negligible because of the depth of water shielding the fuel and the decay of the more active isotopes. Routine exposure increase resulting from radionuclide concentrations in the spent fuel pool water should not be significant since the fuel pool filter-demineralizer units are capable of maintaining the design pool water cleanliness. The concentrations of airborne radionuclides in the spent fuel pool area result mainly from the most recently discharged batch of fuel and will decrease rapidly after refueling. Therefore, only a negligible increase, if any at all, in the spent fuel pool work area is expected as a result of the increased number of assemblies stored in the pool. The only significant increase in routine operational exposures foreseeable is the possible increase in frequency of backwashing the fuel pool filter-demineralizers and the associated man-rem exposures of these operations. A very conservative estimate would be that the spent resin volume would double. Based on past experience, this would result in an addition of 0.3 man-rem/year to the total routine operational exposure for Hatch 1 and a similar addition for the operational exposure for Hatch 2. 2

10.2 Noble Gases

Krypton-85 is released to the pool water and subsequently to the refueling floor atmosphere from the leaking fuel assemblies. For normal operating conditions, most of the krypton comes from the most recently discharged batch of fuel. After the most recent batch has cooled in the pool for 12 months, the pressure buildup in a fuel pin which causes the release of krypton has become very small. Thus, the increase in krypton-85 activity attributed to the increase in spent fuel pool storage capacity will be small compared to the total quantity of all noble gases released from the pools and negligible when compared to the annual plant noble gas releases. Despite the presence of some defective fuel bundles in the Unit 1 pool, krypton-85 activity levels in the refueling floor ventilation exhaust are below the minimum detectable level of approximately 10^{-8} $\mu c/cc$.

10.3 Gamma Isotopic Analysis for Pool Water

Hatch Unit 1 has undergone three refuelings. Typical radioactive isotope concentrations in the Unit 1 spent fuel pool water are presented in Table 10-1 at various dates.

10.4 Dose Levels Over and Along Sides of Pool

Dose surveys at Hatch Unit 1 indicate that after every refueling outage the radiation field over the pool surface has returned to an apparent equilibrium of approximately 1 mr/hr. Local areas show 4 mr/hr (e.g., around the fuel grapple).

Measurements taken during the May 1979 refueling outage show that the radiation levels along the sides and center of the pool are essentially the same (approximately 2 mr/hr). This indicates there has been no significant crud build up around the sides of the pool and the radiation levels are as low as reasonably achievable.

Crud build-up along the sides of the Unit 1 pool has not been significant in the past. The spent fuel movements necessary for the Unit 1 modifications will involve removing approximately 424 fuel assemblies from the Hatch 1 pool. These assemblies will be transferred to the Unit 2 pool through the transfer canal. The movement of the 424 assemblies represents fewer movements than a full core off-load which was done during the 1979 refueling outage on Unit 1. Exposure levels above the pool during the 1979 refueling outage did not exceed the 3 mrem/hr maximum level used in estimating man-rem dose accumulations for the overall proposed modification presented in Section 6. Additional crud may be dislodged during removal of the old racks; however, this is not expected to increase the dose rate over the pool. Waterborne crud in the pool will be continually removed by the spent fuel pool filter-demineralizers. An underwater vacuum may be used to aid this clean-up if necessary. 2

10.5 Airborne Radioactive Nuclides

Air samples taken from the Unit 1 refueling floor atmosphere during and after each refueling showed activity levels below the lower level of detection. Storage of additional fuel is not expected to increase the airborne activity on the refueling floor since the major contribution of airborne activity is attributed to the most recent batch of spent fuel that is placed in the pool.

10.6 Radiation Protection Program

The Radiation Protection Program is described in Section 12.5 of the Hatch 2 FSAR. This program will be adhered to during the removal of the old racks and installation of the new racks.

10.7 Disposal of Present Spent Fuel Racks

There are at present 42 aluminum racks in the Unit 1 pool and 56 in Unit 2. Each rack weighs about one ton. Presently, there is no fuel stored in the Unit 2 spent fuel pool. The racks removed from Unit 2 will be prepared and stored in the warehouse for future sale or use. The racks and seismic restraints from the Unit 1 pool will be decontaminated, crated, and shipped offsite to a licensed burial location. This represents an estimated volume of 10,000 cubic feet of contaminated materials. A reasonable effort will be made to limit personnel exposures to as low as reasonably achievable during this work. 2

10.8 Impact on Radioactive Effluents

The spent fuel pool has its own filter/demineralizer system, and under normal circumstances the spent fuel pool water is not transferred to the liquid radwaste system. Therefore, no increase in liquid effluents from the plant is anticipated as a result of the proposed pool modifications.

The spent fuel pool leakage collection system is comprised of embedded stainless steel channels behind the stainless steel liner plate, which provide interconnected drainage paths for the pool walls and slab. The leak off connections from the channels drain through open funnels into drain lines, as shown in Unit 2 FSAR Figure 9.1-4, that direct the flow to the reactor building dirty radwaste sumps located in the southwest and southeast corner rooms. The sumps, pumps, level instrumentation and system operation are discussed in the Unit 2 FSAR Sections 9.3.3.2 and 9.3.3.3. Liner leaks can be visually observed at the open funnels and can be monitored by observing the frequency and duration of the sump pump runs. Presence of large leaks would be annunciated in the control room by level switches on the sumps. The design features described above for Unit 2 are applicable for Unit 1. 2

In addition, abnormal spent fuel pool water level alarms are provided in the control room. Level switches are also provided on the skimmer surge tanks which will initiate alarms for high, low, and low-low surge tank levels. A low level alarm can be an indication of a leak in the system.

There has not been any leakage from the spent fuel pool in the past on Unit 1; however, should leakage occur, it can be detected through an increase of the make up water, a visual inspection of the liner leak off connections, and/or unusual frequency of operation of the sump pumps.

RESPONSE TO NRC QUESTIONS

CONTAINED IN THE NRC'S

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QUESTION 1

Provide an estimate of the man-rem exposure that will be received during the removal and disposal of the old racks from Unit 1 and installation of the new high density racks. The estimate should include the number of workers involved in each phase of the operation including divers, if any; the duration of the operation; the exposure rate during each phase of the operation; and the total man-rem received by all workers involved. Relevant experiences may be cited. Discuss how the estimation of the man-rem exposure above would be effected if the Hatch-2 SFP should become contaminated prior to its modification.

RESPONSE:

The response has been incorporated into Section 6.0.

QUESTION 2

Provide an estimate of the annual man-rem expected from all operations in the SFP area, including refueling, based on the fission and corrosion product concentrations in the spent fuel pool water indicated in Table 10-1 of your July 9, 1979 submittal. Although Section 10.1 states that "the increase in the SFP storage capacity is not expected to appreciably affect the annual man-rem dose," estimate the increase of this man-rem dose as a result of the modification of Hatch 1 and 2 since the modification should increase the radioactive source inventory in the SFP at some time in the future.

RESPONSE:

The response has been incorporated into Section 10.1.

QUESTION 3

Provide the estimated volume of contaminated material (e.g., spent fuel racks, seismic restraints) expected to be shipped from the plant because of the pool modification to a licensed burial site.

RESPONSE:

The response has been incorporated into Section 10.7.

QUESTION 4

Discuss in some detail the impact of the proposed pool modifications on radioactive liquid effluents from the plant. Include a discussion of the pool leak collection system, pool leak detection system, and history of leakage from the pool.

RESPONSE:

The response has been incorporated into Section 10.8.

QUESTION 5

Provide the estimated failed fuel fraction for each fu-l cycle at Hatch-Unit 1.

RESPONSE:

The following table provides the number of fuel failures and the number of 7x7 bundles in the core for cycles 1-3 (sipping data) and cycle 4 (estimated).

Cycle	No. of 7x7 Failures	7x7 Bundles in Core
1	2	560
2	4	468
3	4	300
4	4 (estimated)	136

The Hatch-1 estimated cumulative failed fuel fraction is therefore 14/560 (0.025). This compares favorably with an industry-wide (G.E. BWR fuel) cumulative failed fuel fraction between 0.1 and 0.15 for 7x7 fuel.

Hatch-1 has not experienced any 8x8 failures through cycle 3. However, it is estimated that the cumulative failed fuel fraction for 8x8 fuel will be less than 0.08.

QUESTION 6

You stated in Section 8 of your submittal dated July 9, 1979 that the design pool bulk water temperature will be above the FSAR design value of 125°F during normal refuelings after the eighth refueling. If the actual expected value of the bulk water temperature, not the design value, may be above the 125°F under realistic conditions, discuss when this will occur during any refueling, for what period of time it will occur, the maximum value of the temperature and the effect of this on releases of radioiodine and tritium from the pools.

RESPONSE:

The response has been incorporated into Section 8.3.3.

QUESTION 7

Discuss the effect of the spent fuel movements during the modifications of the Hatch 1 pool on the amount of crud in the pool water and the radiation levels in the vicinity of the pool during the pool modification.

RESPONSE:

The response has been incorporated into Section 10.4.

QUESTION 8

Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC staff.

RESPONSE:

As discussed in Section 11.0, such information has previously been provided in Reference 6 which was prepared in response to an NRC letter dated May 17, 1978 pertaining to Task Action Plan for Category A Technical Activity No. A-36, "Control of Heavy Loads Near Spent Fuel".

QUESTION 9

Provide a list of all objects that are required to be moved over or near the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram or description of the transfer path utilized and the frequency of movement.

RESPONSE:

Refer to the response to Question 8.

QUESTION 10

For our evaluation of the difference between the maximum calculated k_∞ of 0.87 given in your submittal and the maximum actual k_{eff} that might occur in the spent fuel pool, the following information should be provided:

a. The quantity and distribution of the uranium-235 in the fuel pool storage lattice calculation for this maximum k_∞;

b. The quantity and distribution of gadolinium-155 and gadolinium-157 in the fuel pool storage lattice calculation for this maximum k_∞;

c. The quantity and distribution of the fission products and actinides in the pool storage lattice calculation for this maximum k_∞;

RESPONSE:

The maximum calculated fuel storage k_∞ of 0.87 was derived using a fuel bundle design which has a k_∞ of 1.35 in the reactor core geometry at 20°C for the uncontrolled state. The design value of $k_\infty = 1.35$ was conservatively selected as an upper bound value for the storage racks since fuel with $k_\infty = 1.35$ would not be able to satisfy the core design requirements for shutdown margin, control rod worth, etc. Since there is a wide range of fuel designs for BWR's, it is not practical to bound all design variables such as U-235 enrichment and distribution, gadolinia concentration and distribution, and fission product inventory. The one common denominator which can be used to characterize the reactivity associated with the various BWR bundle designs is the infinite lattice multiplication factor calculated for reactor core geometry.

In response to questions 10a and 10b, reload licensing topical report NEDO-24011 gives the U-235 and gadolinia concentrations and distributions for all GE BWR reload bundle designs. Furthermore, the maximum k_∞'s for all bundle types are presented in Section 3.3.?.1.4 of NEDO-24011. As can be seen from these results, the maximum k_∞ value for all fuel types is 1.26.

In calculating these k_∞ values, the U-235 and gadolinia concentrations and distributions, along with the distribution of heavy metal isotopes, fission products, and actinides (question 10c) are included as described in the lattice physics topical report NEDO-20913. The ability of these lattice methods to perform the isotopic burnup evaluations has been verified and presented in the lattice physics methods verification topical report NEDO-20939.

The licensing criterion for fuel storage, as stated in the Standard Review Plan, is $k_{eff} = 0.95$. A value of 0.87 was calculated for a fuel bundle with $k_\infty = 1.35$, whereas, the maximum k_∞ as reported in NEDO-24011 is 1.26. This results in a design allowance of ~0.16 - 0.17 Δk between the standard reload fuel designs and the licensing criterion of 0.95.

QUESTION 11

On page 7-3 of your submittal you state, "The results in Table 7-1 show that the nominal pitch (Case 2) has a higher k_∞ result than the minimum pitch case (Case 6)." Since the statistical error bounds given on Table 7-1 do not preclude the opposite conclusion, and since all other calculations for similar storage lattices show the opposite to be true, provide a justification for this conclusion.

RESPONSE:

Item d. on page 7-3 has been restated to correct this discrepancy.