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Detroit Edison

A DTE Energy Company

October 19, 2000
NRC-00-0065

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555-0001

References: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

2) Detroit Edison letter to NRC "Proposed Technical Specification Changes (License Amendment) – Design Features/ Fuel Storage (Technical Specification 4.3) And Programs and Manuals/High Density Spent Fuel Racks (Technical Specification 5.5.13)", dated November 19, 1999

Subject: Response to Request for Additional Information on Technical Specifications Change Request Related to Spent Fuel Pool Expansion at Fermi 2 (TAC No. MA7233)

On June 16, 2000, Detroit Edison received a set of NRC questions pertaining to the Criticality, Control and Handling of Heavy Loads and Thermal Hydraulic areas of the proposed Technical Specifications change request to increase the capacity of the Fermi 2 Spent Fuel Pool (Reference 2). On August 23 and August 31, 2000, teleconferences between Detroit Edison, NRC, and Holtec International staffs were conducted to discuss these questions. Enclosed is Detroit Edison's response to the questions.

A001

A commitment is made to revise Fermi 2 Operations Conduct Manual Procedure, MOP 13 to incorporate the 60 hours minimum hold time before moving fuel assemblies from the reactor vessel to spent fuel pool. This procedure revision shall be completed with implementation of Technical Specifications change.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

William T. O'Connor

Enclosure

cc: D. S. Hood
M. A. Ring
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, WILLIAM T. O'CONNOR, JR., do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

William T. O'Connor Jr.
WILLIAM T. O'CONNOR, JR.
Vice President - Nuclear Generation

On this 19th day of October, 2000 before me personally appeared William T. O'Connor, Jr., being first duly sworn and says that he executed the foregoing as his free act and deed.

Karen M. Reed-Ockerman
Notary Public

KAREN M. REED-OCKERMAN
Notary Public, Monroe County, MI
My Commission Expires Sep. 2, 2003

ENCLOSURE TO
NRC-00-0065

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON TECHNICAL SPECIFICATION CHANGE REQUEST RELATED TO SPENT FUEL POOL EXPANSION AT FERMI 2

Responses to Fermi 2 RAIs from Reactor Systems Branch (SRXB)

Question 1:

On page 2 of the submittal, in the last sentence of the 2nd paragraph, it is stated that the current configuration allows for storage of up to 2383. How does one come up with 2383?

Response to Question 1:

The following summary table provides the details of existing storage capacity in Fermi 2 Spent Fuel Pool (SFP). The 2383 value excludes defective fuel rack E15, and two additional cells which are used for the Boraflex surveillance program.

EXISTING RACK AND STORAGE CAPACITY SUMMARY		
Existing Rack Layout		
Rack Description	Size	Total Cells
E1 through E10	13x13	169x10 = 1690
E11	9x12	108
E12 through E14	13x13	169x3 = 507
GE Racks	2x10	20x4 = 80
Boraflex Surveillance Cells	Variable locations	-2
Current Storage Cell Total		**2383**

Question 2:

Modern BWR fuel assemblies are being loaded with Gadolinia concentrations in excess of 5 weight percent Gd_2O_3 in well over 10% of the fuel rods in an assembly. The information supplied in the submittal only refers to Gadolinia loading at much lower levels. The staff has performed some audit calculations that show significant differences in nuclide concentrations when comparing two assemblies (both assemblies have the same initial enrichment with different Gadolinia loading) burned to the point where they have equal K_{inf} in the standard cold core geometry. How does the Holtec methodology account for these types of spectral effects and their effect on reactivity?

Response to Question 2:

The Holtec analysis includes the slightly higher concentrations of nuclides (such as illustrated in the figure attached to the RAI) that result from the spectrum hardening due to Gadolinia. The Holtec methodology for criticality analyses of BWR systems was developed specifically to conservatively bound all fuel currently in use as well as future designs that may be reasonable modifications of the current fuel (i.e., higher enrichment and increased number and loading of fuel rods containing Gadolinia. To assure conservative results in the criticality analysis, a bounding value for enrichment (5%) was used in conjunction with lower values of Gadolinia loading. This is conservative because (1) that enrichment is higher than that currently in use and (2) any Gadolinia loading greater than that identified in the analysis (which would be expected for 5% enriched fuel) would result in a lower reactivity. Although the Licensing support document, HI-992154, analyzed only fuel up to 5% Gadolinia (see figures 4.4.1, 4.4.2, and 4.4.3 of HI-992154) the analysis includes the slightly higher concentration of nuclides, as stated above, that result from the spectrum hardening due to Gadolinia.

The major effect of higher Gadolinia loading is that it absorbs more neutrons, which reduces the peak reactivity over burnup and shifts the point of peak reactivity out to a higher burnup, as shown on figure 4.4.1 of HI-992154. The greater amount of Gadolinia also hardens the neutron spectrum and can tend to produce an opposing effect, as indicated in the illustration attached to the RAI. This is, however, a much smaller effect. Figure 4.4.4 of HI-992154 shows the decrease in reactivity of the rack as the Gadolinia increases. The trend to ever lower peak reactivities would continue at higher Gadolinia loadings.

The spent fuel storage racks for Fermi 2 are designed to safely accept fuel up to 5% average enrichment whose highest planar k_{inf} in the SCCG is 1.33. This limiting k_{inf} could only be reached with a low Gadolinia content. At higher Gadolinia loadings, a k_{inf} in the SCCG of 1.33 could not be achieved and the reactivity of the racks would be conservatively lower than 0.95.

Responses to Fermi 2 RAIs from Plant Systems Branch (SPLB)
Control and Handling of Heavy Loads

Question 1:

1. Figure 1.1.1 of the Holtec Report HI-992154 shows the placement of a 1-inch thick plate on top of the pool liner in the cask area.

   a. Does the cask impact the freedom of Holtec Racks A, B, and H in the Campaign II Rack Layout (Figure 1.2.2) to slide during a seismic event?

   b. Section 7.5.2 states that the distance from the baseplate to the liner is 5.5 inches. Please clarify this given the information provided in Table 3.1.1, which indicates that this distance will be 7.5 inches (nominal).

Response to Question 1(a):

The 1" thick plate on the top of the liner is the cask pad. Its location in the pool will have no effect on the ability of any of the neighboring racks to slide, because each of the support pedestals for each rack is positioned on a 2" thick bearing pad. The maximum rack movement, as predicted by the supporting analysis, is 0.83" (at the top of the rack). Any sliding of the racks will remain within the perimeters of the supporting bearing pads.

Response to Question 1(b):

The distance from the liner to the baseplate is 7.5", as detailed in Table 3.1.1. This value includes the nominal dimension of the rack foot (5.5") plus the thickness of the supporting bearing pad (2"). The reference to this distance as 5.5" in section 7.5.2 is incorrect, because it does not include the thickness of the bearing pad. However, in the context of this section, the dimension was referenced with respect to the deep drop analysis. It is used in explaining that any deformation to the rack baseplate caused by a dropped fuel assembly was less than the distance between the baseplate and the liner, concluding that no damage occurred to the liner in this case. Using the corrected distance of 7.5", the safety margin on this conclusion is magnified.

Question 2:

Figure 1.2.1 of the Holtec Report HI-992154 shows the placement of Holtec Rack A in the cask area during Campaign I, then moved during Campaign II (Figure 1.2.2). Why is this rack not placed in its final location initially during Campaign I, which would eliminate the need to relocate this rack during Campaign II?

Response to Question 2:

Rack A is shown in a temporary location in Campaign I because it will be installed only for an emergency core offload situation. Presently, it is being delivered and stored at the site as part of the initial campaign. In order for rack A to be installed in its final location, as shown in Figure 1.2.2 of the licensing report, existing piping is required to be removed. This work requires diving activities and is not planned until Campaign II.

Question 3:

The proposed reracking increased the amount of storage cells in the pool and places racks in areas of the pool which did not previously have racks. Please verify that the fuel handling system will have sufficient travel to access all cells of the new spent fuel racks.

Response to Question 3:

The refueling bridge main hook will not access rack cells on the easternmost row of cells in the racks installed along the east wall. It should be noted that such racks are not to be installed until Campaigns II and III. Campaign I is not affected by this limitation. In order to install spent fuel bundles in these affected cells, either an offset tool will be used in conjunction with the main hook or the frame- or the monorail-mounted hook will be used to handle the bundles.

Question 4:

Section 10.2 of the Holtec Report HI-992154 describes the lift rig which will be used to place the racks into the pool. Will the same fixture be used for the asymmetric racks (Holtec Racks E, F, and R)? If so, are any special precautions needed for handling these racks? What are they?

Response to Question 4:

The same lift rig is used for all new racks. In order to address the issue of asymmetrical racks, large turnbuckles are used in each of the four load paths leading from the four lift rig eyepads to aid in leveling a hanging rack in which the lift rig is offset from the rack's center of gravity.

Question 5:

Section 3.4 of the Holtec Report HI-992154 states that Holtec Racks E and F will contain several dual purpose cells. While not initially for fuel storage, these cells can be converted into storage cells by installing a cruciform insert. Please describe the process, equipment, fixtures, and inspections required to remotely assemble the insert into the rack.

Response to Question 5:

A special tool is supplied to lift the cruciform inserts. The tool is suspended from the overhead building crane. The cruciform insert is installed in its target cell and then the special tool is remotely disengaged from the insert. Aside from a visual inspection to ensure that the cruciform insert is seated on the rack baseplate by comparing the cruciform elevation to that of the rack cell walls, no other inspections are required.

Question 6:

Section 3.4, Module Layout, of Holtec Report HI-992154 states that Holtec Racks B and G are designed to support an overhead platform, which can store miscellaneous objects up to 5 tons total weight. Additional limitations regarding storage of items are also provided in Section 6.3.2. What procedures or operating restrictions will be prepared to ensure these limitations are complied with? Which, if any, of the previously completed Holtec reracking projects (Table 1.1.2) utilize the overhead platform concept?

Response to Question 6:

The placing of items on or removing items from the Holtec Overhead Platform (HOP) will be governed by Fermi 2 work control procedures as well as hoisting, rigging and load handling procedures. These procedures require an engineering analysis when loads are transported over spent fuel. These procedures also require that the work package governing the work activity contains a plant impact statement, supporting data and / or drawings. The impact statement addresses the impact on plant systems, structures and components of the proposed work activity.

A HOP is currently in use at Niagara Mohawk's Nine Mile Plant.

Question 7:

Please verify that the personnel training referenced in Section 10.2, Installation, of Holtec Report HI-992154 will be in accordance with the provisions of NUREG-0612, Section 5.1.1(3), which specifies training shall be conducted with the provisions of ANSI B30.2-1976.

Response to Question 7:

As stated in the licensing report, the crew members will be trained on the use of the lift rigs, upender, and other aspects of the rerack project. This training is not governed by any aspects of NUREG-0612. This is additional training provided by Holtec as part of the defense-in-depth approach suggested by NUREG-0612. However, training of crane operators with respect to crane operation shall be completed by Detroit Edison and, as stipulated in NUREG-0612, complies with the requirements of ANSI B30.2-1976.

Question 8:

Section 10.2, Installation, of Holtec Report HI-992154 states that the cranes and lifting devices used will be given a preventive maintenance checkup and inspection per Fermi 2 plant procedures. Please verify that these crane procedures will be in accordance with NUREG-0612, Section 5.1.1(6), which specifies that the inspections are to be in accordance with ANSI B30.2-1976.

Response to Question 8:

Detroit Edison will provide preventive maintenance of the cranes prior to commencement of site activities as specified by NUREG-0612. As stipulated in the NUREG, the maintenance provided by Detroit Edison complies with ANSI B30.2-1976.

Question 9:

Section 10.2, Installation, of Holtec Report HI-992154 states that a combination of the single-failure proof cask handling crane and the 5-ton auxiliary crane will be used for the final placement of the racks.

a. Please explain when the 5-ton crane will be used, given that the weight of the proposed new racks specified in Table 3.4.1 (with the exception of Holtec Racks E and F) exceed 5-tons weight.

b. Page 4 of Attachment 2 to NRC-99-0084 states that a new temporary hoist and rack lift rig will be used to lift and suspend the racks from the bridge of the reactor crane. What is the capacity of the additional temporary hoist which will be used to prevent submergence of the cask handling and auxiliary crane hooks? What are the design qualifications for the temporary hoist and rack lift rig?

Response to Question 9(a):

The actual installation and handling of the racks shall be done by the single failure proof cask handling crane. The 5-ton auxiliary hook may be utilized for the removal of the lift rig after rack installation or handling of the long-handled rack leveling tool used in the final leveling of the rack after installation.

Response to Question 9(b):

The rack installation plan calls for the main hook to be used and submerged in the pool, thus making the need for a temporary hoist unnecessary. However, should it be necessary to use the temporary hoist, it can be used with the lift rig for rack installation. The lift rig, as required by NUREG-0612 for special lifting devices, is designed per the guidelines of ANSI N14.6 – 1978. With respect to the temporary crane, NUREG-0612 stipulates that lifting devices that are not

specially designed shall either provide redundancy or be rated for twice the load (static + dynamic). In complying with the latter of the two options, the temporary hoist shall be rated for a minimum of 37.5 metric tons.

Question 10:

Section 1.3 (page 1-5) of Holtec Report HI-992154 states that Section 10 provides an outline of the construction services for Campaign I. A review of this section does not make this distinction. Are the details pertaining to the construction services limited to only the first campaign, or are all three campaigns described? If not, what will be different for the subsequent campaigns?

Response to Question 10:

Section 10 is intended to be a summary of the complete reracking evolution for all of the planned campaigns. However, the removal of racks does not have a separate heading, but is detailed within subsection 10.9, entitled "Radwaste Material Control".

The basic handling of new racks and existing racks is the same, with lift rigs, cranes, and safe load paths all required for each of the evolutions and, in most cases, is the same regardless of the rack being handled.

Question 11:

Section 7.2.1.3 of Holtec Report HI-992154 described the results of a dropped gate on a loaded spent fuel rack and states that this accident will bound the drop of an overhead platform due to the weight of the gate being much larger (9479 lbs. vs. 1460 lbs). Please provide additional information regarding the consequences of a dropped platform. To what extent will fuel rod damage occur if the four legs of a dropped platform impact four fuel assemblies?

Response to Question 11:

The consequences of a dropped overhead platform onto a fuel assembly result in no rod rupture even with the assumption that the impact of the dropped overhead platform is sustained by one fuel assembly alone. The following table shows a comparison of the impact energies calculated for both the fuel pool gate drop and the overhead platform drop.

Drop Information	Types of Drop onto Stored Fuel Assembly	
	Pool Gate	**Overhead Platform**
Weight of Impactor (lbs)	9479.78	1460
Drop Height (inches)	30	480 (conservative)
Frontal Area of Impactor (in^2)	388.5	50×50 (conservative)
Impact Velocity (in/sec)	133.3	106.7
Impact Energy (lbf*in)	2.18×10^5	2.151×10^4

The above comparison of the impact energies indicates that the overhead platform drop is well bounded by the gate drop. The calculation of the impact energy for the overhead platform drop is highly conservative due to the fact that the overhead platform is assumed to drop from 40 feet of height and on top of a single fuel assembly, which is more severe than four legs of a dropped platform impacting four fuel assemblies.

Question 12:

Section 10.5.1 of Holtec Report HI-992154 states that it may be necessary to shut off SFP cooling during specific installation steps (i.e., bearing pad elevation measurements). Please provide additional explanation regarding why the operation of the SFP cooling system would impact these processes?

Response to Question 12:

In some areas of the pool, the flow of the water from the cooling system may cause a hanging rack to swing, thus making it difficult to install in its design location. Also, suspended plumb bobs are used to confirm the rack levelness after installation. Again, the water flow will not allow such a small weighted item to be suspended motionless, a requirement necessary to ensure accurate measurements.

Question 13:

Section 10.6 of Holtec Report HI-992154 states that the fuel will be shuffled in the pool in accordance with the Technical Specification requirements. Please specify which requirements are referenced and ensure that this does not present a misplaced fuel assembly scenario that was not analyzed for. In particular address the following questions:

a. As stated in proposed Technical Specification 4.3c, there are four types of racks, including two high-density racks with different poison materials (Boral and Boraflex). Are there any restrictions on placement of fuel in these racks?

b. In the event of Boraflex degradation, how are the restrictions on fuel placement implemented (Tech Specs, plant procedure, etc)?

c. What specific Technical Specifications are referred to in Section 10.6?

d. If there are fuel placement restrictions, is the misplacement of a SFA in the incorrect rack bounded by the misplaced fuel assembly discussion contained in Enclosure 2 to NRC-99-0084 Significant Hazards Consideration?

Response to Question 13a:

No. There are no restrictions on the placement of fuel in the racks when shuffling fuel. The Technical Specifications do impose a limit on the reactivity of a fuel assembly (k-infinity in the standard cold core geometry) of 1.31, but this is a limit imposed on all of the racks universally and is more applicable to the design of new fuel.

The only exception is a physical limitation associated with the Defective Fuel Rack. If a fuel assembly is to be placed into the Defective Fuel Rack, then it must be placed into a defective fuel storage container. (The Defective Fuel Rack consists of a series of cylindrical tubes that would house the cylindrical defective fuel storage containers into which defective fuel assemblies would be placed.)

Response to Question 13b:

We have no restrictions on fuel placement due to Boraflex degradation. Fermi has a Boraflex monitoring program, as discussed in our response to Generic Letter 96-04. We don't believe our Boraflex degradation to be significant based on results of the monitoring program.

Response to Question 13c:

The following are Technical Specifications associated with equipment operability for handling of irradiated fuel in Secondary Containment (shuffling fuel in the spent fuel pool):

- T.S. 3.3.6.2 Secondary Containment Isolation Instrumentation (per T.S. Table 3.3.6.2-1)
- T.S. 3.3.7.1 Control Room Emergency Filtration System Instrumentation (per T.S. Table 3.3.7.1-1)
- T.S. 3.3.8.1 Loss of Power Instrumentation
- T.S. 3.6.4.1 Secondary Containment
- T.S. 3.6.4.2 Secondary Containment Isolation Valves
- T.S. 3.6.4.3 Standby Gas Treatment System
- T.S. 3.7.3 Control Room Emergency Filtration System
- T.S. 3.7.4 Control Center Air Conditioning System
- T.S. 3.7.7 Spent Fuel Storage Pool Water Level
- T.S. 3.7.8 Emergency Diesel Generator Service Water System
- T.S. 3.8.2 Electrical Power Systems – AC Sources-Shutdown
- T.S. 3.8.3 Electrical Power Systems – Diesel Fuel Oil and Starting Air
- T.S. 3.8.5 Electrical Power Systems – DC Sources-Shutdown
- T.S. 3.8.6 Electrical Power Systems – Battery Cell Parameters
- T.S. 3.8.8 Electrical Power Systems – Distribution Systems-Shutdown

Response to Question 13d:

As discussed in Question 13a above, there are no fuel placement restrictions imposed on the fuel storage racks.

Question 14:

Table 10.2.1 of Holtec Report HI-992154 states that safe load paths have been defined for the movement of the racks to minimize the potential impact on safe shutdown equipment. The Holtec Report does not provide additional detail regarding this analysis. Describe the safe load paths which have been defined and which safe shutdown equipment may be potentially impacted during the raising of the racks to the refuel deck and subsequent placement into the pool.

Response to Question 14:

Safe load paths for rack handling on the elevation of the SFP operating floor shall mimic the plant safe load paths for heavy loads for this floor elevation. However, the safe load path defined for heavy loads is presently laid out for the transit of a cask from the hatchway to the cask pit in the pool. Because new and old racks are located throughout the pool, entry and removal locations of new and old racks shall be over the south, west, and north pool walls. In summary, the safe load paths for all racks, whether being removed or installed, shall mimic the safe load path for casks between the hatch and the pool. However, once the pool perimeter is reached, new racks will travel as necessary to reach their respective installation points in the pool along one of the three previously mentioned pool walls.

For racks being removed from the pool, the rack will be moved to a point along one of the desired pool walls (south, west, or north), raised from the pool, and moved to a point on the operating floor on the west side of the pool to align with the established safe load path leading from the west side of the pool to the hatchway. While no safe shut down equipment is in the load path, the following equipment is in the vicinity of the load path:

(1) Division 2 Primary Containment Monitoring System instrument rack.
(2) Division 2 Primary Containment terminal box.
(3) Division 1 Non-interruptible Air Supply lines to the inboard RB truck bay doors.
(4) Division 1 NIAS railroad air lock door seal volume chamber

Thermal Hydraulic Considerations

Question 15:

The SFP cooling portion of the Fuel Pool Cooling and Cleanup (FPCC) system is designed to maintain the SFP temperature below 150°F during refueling outages or below 125°F during plant power operation following a refueling outage. The decay heat removal Residual Heat Removal (RHR) system is used to serve as a back-up system to the FPCC system and to provide supplement SFP cooling in the event that the FPCC system cannot maintain the SFP temperature below 125°F. Please provide the following additional information regarding the SFP cooling systems:

a. NRC SRP 9.1.3, Section III.1.d, states that a single active failure should be assumed for the normal maximum heat load with normal cooling system in operation. Since full core offload is a current practice at the Fermi 2 plant during normal refueling, please discuss what single active failure of the SFP cooling systems was considered in establishing the temperature limit of 150°F during a planned full core off-load operation. Please also discuss how many trains of FPCC system and RHR system are required to the operable and available for SFP cooling prior to a planned full core offload.

b. On page 5-5 of Holtec Report HI-992154, Holtec states that the current Fermi 2 operating procedures have provisions to require the alignment of one train of the RHR to cool the SFP once the 125°F point is reached. Does this mean that one train of RHR is allowed to be aligned to the SFP even before the completion of the core discharge, if the SFP temperature exceeds 125°F? Is this alignment incorporated in the Fermi 2 operating procedures or technical specifications? Are there any operating procedures or resolutions that could preclude realignment of RHR from reactor cooling to supplement FPCC system during the refueling operation?

Response to Question 15(a):

Since the current practice at Fermi 2 is to perform partial core offloads while retaining the capability to perform a full core offload, a single failure analysis was performed for the maximum heat load during normal partial-core offload. Per NRC SRP 9.1.3, Section III.1.d single active failure need not be considered for abnormal full core offload.

During a normal partial-core offload with bounding decay heat load and cooling water conditions, one RHR division will be aligned to cool the SFP. The worst-case single active cooling system failure, with respect to the SFP, would be the loss of a division of RHR aligned to cool the pool. Credit was taken for placing the two trains of FPCC system back into service, with their respective demineralizers bypassed, to cool the pool.

Under this postulated single-active failure, the maximum pool bulk temperature is calculated to be 165°F. The maximum local temperature and maximum fuel cladding temperature are conservatively calculated to be 193°F and 221°F, respectively.

Both of these values are still well below the local saturation temperature of 238°F. Hence, the spent fuel remains adequately cooled.

The analysis also concludes that since the evaporation rate under boiling is less than the minimum available make up rates, as described in HI-992154 section 5.8.2, pool water level is maintained and the spent fuel is adequately covered with water.

The effect of the bulk temperature transient (<15°F above 150°F) for a period of less than 8 days associated with the single-active failure condition has also been considered in the SFP structure evaluation to determine that the structural safety margins continue to meet the applicable codes (American Concrete Institute, Standard Review Plan) acceptance criteria.

The FPCC system has no operability requirements in Technical Specifications, however it is the preferred method of cooling SFP. Technical Specifications 3.9.7 and 3.9.8 require at least one train of RHR to be operable while the plant is in mode 5 and the reactor cavity is flooded. Therefore, if needed, one division of RHR could be employed to cool the spent fuel pool.

Response to Question 15(b):

The alignment of RHR to the SFP cooling assist mode is based solely on the temperature of the water in the SFP. This temperature is continuously monitored by plant instrumentation, control room recorder G41R600. Should the SFP water temperature exceed the alarm set point a control room alarm is received. The operators take action based on the alarm response procedure, currently designated 2D9. There is no technical specification addressing SFP water temperature.

The plant operating procedure for the RHR system, currently designated 23.205, provides specific prerequisites and steps needed to align one division of RHR to the SFP cooling. Procedure 23.205 directs operators to comply with technical specifications for ECCS - Operating and Shutdown (3.5.1 and 3.5.2 respectively), RHR Shutdown cooling - hot and cold shutdown (3.4.8 and 3.4.9 respectively), suppression pool cooling and spray (3.6.2.3 and 3.6.2.4 respectively), and RHR – high and low water levels (3.9.7 and 3.9.8 respectively). Technical Specifications 3.9.7 and 3.9.8 require at least one train of RHR to be operable while the plant is in mode 5 and the reactor cavity is flooded. Therefore, if needed, one division of RHR could be employed to cool the spent fuel pool.

Question 16:

With regard to the decay heat calculation for an emergency (unplanned) full core off-load, SRP 9.1.3, Section III. 1.h.iii, recommends that the SFP cooling system should have the capacity to remove the decay heat from one full core at equilibrium conditions after 150 hours of decay and one refueling load at equilibrium conditions after 36 days of decay. On page 5-5 of the Holtec Report (HI-992154), Holtec stated that a minimum decay time of 12 months was assumed for the previously discharged fuel assemblies for the scenario of an unplanned full core discharge. Please provide justifications for the above deviation from the guidance described in the SRP.

Response to Question 16:

The current Fermi 2 design-basis unplanned full-core discharge specifies a minimum decay time of 12 months for the previously discharged fuel assemblies. To evaluate the peak SFP bulk temperature using the SRP recommended minimum decay time of 36 days, Detroit Edison has performed an analysis of this condition. The results of this analysis show that the SFP bulk temperature will not exceed 142°F, well below the required acceptance criterion of no bulk boiling as stated in the SRP 9.1.3, Section III.1.h.iii.

It is worth noting that the results of the supplemental analysis are within 1°F of the unplanned full-core discharge scenario evaluated in Holtec Report HI-992154.

Question 17.1:

In Table 5.8.1 of the Holtec Report HI-992154, Holtec indicates that for the planned full core off-load scenario, the calculated SFP bulk temperature of 140.55°F peaked at the time of 160 hours after reactor shutdown with its corresponding decay heat load of 41.84 x 10^6 Btu/hr. Please provide the following information:

a. Fuel assemblies in-reactor vessel hold time prior to being discharged.

b. Has the fuel assemblies in-reactor vessel hold time been incorporated in the Fermi 2 Technical Specifications (TS), administrative control, or plant operating procedures?

c. The decay heat generated from each batch of the previously discharged spent fuel assemblies (SFAs) and from the freshly discharged full core in the SFP.

d. Has the single active failure of the SFP cooling system been considered in calculating the above peak SFP bulk temperature of 140.55°F?

Response to Question 17.1(a):

The SFP bulk temperature analysis performed for the full core offload scenario utilizes the minimum in-core hold time of 60 hours from Table 5.4.1. Use of the minimum hold time maximizes the decay heat load in the SFP.

Response to Question 17.1(b):

The requirement for a minimum hold time of 60 hours before moving fuel assemblies from the reactor vessel will be placed into plant procedure MOP 13 – Conduct of Refueling and Core Alterations.

Response to Question 17.1(c):

Table A, attached to these RAI responses, presents the decay heat load of each previously discharged fuel batch and the full core decay heat at 160 hours after shutdown for the full core offload scenario.

Response to Question 17.1(d):

The SFP peak bulk temperature of 140.55°F does not include the effects of a single active failure of the SFP cooling system. The full-core offload is not the normally practiced refueling procedure at Fermi 2, so there is no requirement to consider the effects of a single active failure. The impact of a single active failure of the spent fuel pool cooling systems for the current practice of partial core offload is addressed on the response to question 15 a.

Question 17.2:

In Table 5.8.1 of Holtec Report HI-992154, Holtec indicates that for the unplanned full core off-load scenario, the calculated SFP bulk temperature of 141.08°F peaked at the time of 159 hours after reactor shutdown with its corresponding decay heat load of 42.37 x 10^6 Btu/hr. Please provide the following information:

a. Fuel assemblies in-reactor vessel hold time prior to being discharged.

b. Has the fuel assemblies in-reactor vessel hold time been incorporated in the Fermi 2 Technical Specifications (TS), administrative control, or plant operating procedures?

c. The decay heat generated from each batch of the previously discharged spent fuel assemblies (SFAs) and from the freshly discharged full core in the SFP.

Response to Question 17.2(a):

The SFP bulk temperature analysis performed for the unplanned full core offload scenario utilizes the minimum in-core hold time of 60 hours from Table 5.4.1. Use of the minimum hold time maximizes the decay heat load in the SFP.

Response Question 17.2(b):

The requirement for a minimum hold time of 60 hours before moving fuel assemblies from the reactor vessel will be placed into plant procedure MOP 13.

Response to Question 17.2(c):

Table B, attached to these RAI responses, presents the decay heat load of each previously discharged fuel batch and the full core decay heat at 159 hours after shutdown for the unplanned full core offload scenario.

Question 17.3:

Table 5.4.1 shows that the minimum in-core hold time is 60 hours. Please clarify whether this minimum in-core hold time of 60 hours is only for the partial core discharge scenario or for all three (partial core, planned full core, and unplanned full core discharge) scenarios?

Response to Question 17.3:

The SFP bulk temperature analyses performed for all offload scenarios (partial core, full core and unplanned full core) utilize the minimum in-core hold time of 60 hours from Table 5.4.1. Use of the minimum hold time maximizes the decay heat load in the SFP.

Question 18:

Discuss the procedures or programs established to monitor and control the SFP water temperatures during planned refueling outages and unplanned full core offload events. Information should include:

a. Discuss how often the water temperature in the SFP will be monitored.

b. Provide the set-point of the high water temperature alarm for the SFP.

c. Discuss the precautions actions (i.e., prohibiting fuel handling or aligning other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature.

d. Information supporting a determination that there is sufficient time for operations to intervene in order to ensure that the temperature limit of 150°F will not be exceeded.

Response to Question 18(a):

Permanently installed plant instrumentation monitors the temperature of the SFP water. The SFP water temperature is displayed on a control room recorder. The Control Room Nuclear Supervising Operator (NSO) is procedurally required to remain aware of the status of all plant equipment. Should the spent fuel pool temperature exceed the alarm set point, alarm response procedure 2D9, provides operator direction on how to proceed.

Response to Question 18(b):

The alarm set point for SFP water temperature is 130°F increasing ± maximum instrument channel inaccuracy of 5°F. Therefore the alarm range is 135 °F to 125°F.

Response to Question 18(c):

The alarm response procedure, 2D9, directs operators to increase the SFP water heat removal rate via the FPCC system in accordance with operating procedure, 23.708. If the SFP water temperature cannot be maintained less than 150°F utilizing the FPCC system, the alarm response procedure, 2D9, directs the operators to align one division of RHR to the spent fuel pool cooling assist mode per procedure 23.205.

Response to Question 18(d):

No specific analysis was performed to define the time available for operator intervention to assure SFP water temperatures would not exceed 150°F. In Holtec Report HI-992154 it was determined that in the event that all cooling to the SFP were lost, the SFP would boil in 4.2 hours. This analysis determined that this was sufficient time for operators to intervene and provide adequate make up water streams to maintain SFP level within normal operating ranges thereby maintaining adequate shielding of irradiated spent fuel.

Question 19:

On page 5-8 of Holtec Report HI-992154, Holtec states that the differential equation on Page 5-6 was solved numerically to obtain the transient thermal response of the SFP. To solve the equation, it is necessary to know either p (heat exchanger temperature effectiveness), or t_o (coolant outlet temperature) for the Qhx term on page 5-7. Please explain how these values are obtained. What SFP temperature was assumed at the beginning of the core discharge operation to calculate the maximum bulk temperatures (i.e., the initial condition to solve this differential equation? Also, what is the SFP water flow rate assumed to be circulated through the RHR heat exchanger?

Response to Question 19:

An equation for calculating temperature effectiveness values for the FPCCU system and RHR heat exchangers was presented on Page 5-7 of Holtec Report HI-992154. The cooling water temperature difference (t_o - t_i) for each of the heat exchangers is calculated using a simple mass-energy balance (i.e., $Q = m \times c_p \times (t_o - t_i)$) and the design-basis performance conditions presented on Page 5-3 of Holtec Report HI-992154.

The initial SFP bulk temperatures for the partial core, full core and unplanned full core offload scenarios were 115.14°F, 115.14°F and 116.14°F, respectively. These temperatures correspond to the steady-state temperatures for each scenario based on their respective previously discharged decay heat loads.

As stated on Page 5-3 of Holtec Report HI-992154, the RHR pump will deliver approximately 3500 gpm for the designed piping configuration.

Question 20:

With regard to the calculated minimum time-to-boil, on page 5-8 of Holtec Report HI-992154, Holtec states that the loss of forced cooling is assumed to occur coincident with the SFP peak decay heat generation; and in Table 5.8.2, Holtec indicated that the calculated minimum time-to-boil is 4.2 hours. Since there is a time-lag between the SFP peak decay heat generation and the calculated peak SFP temperature, please provide the following information:

a. The calculated minimum time-to-boil with the assumption that the loss of forced cooling occurs coincident with the calculated peak SFP temperature.

b. Discuss how the duration of the calculated minimum time-to-boil will be sufficient for the operators to intervene.

c. Discuss the plant operating procedures which require to provide makeup water for boil off from various sources.

Response to Question 20(a):

Table C, attached to these RAI responses, presents the calculated minimum time-to-boil with the assumption that the loss of forced cooling occurs coincident with the calculated peak SFP temperature. The time-to-boil values presented in Table C are slightly larger than those presented in Table 5.8.2 of Holtec Report HI-992154.

Response to Question 20(b):

The operators' initial intervention following a loss of forced cooling would be to align an adequate makeup water supply prior to the onset of bulk SFP boiling to ensure that the SFP

water level is maintained. The minimum time available for aligning adequate makeup is 4.2 hours, as stated in Holtec Report HI-992154. There are several makeup water sources available that could be aligned in less than *one* hour, including:

i. up to 2000 gpm from the RHR service water system
ii. up to 100 gpm from the condensate storage system
iii. up to 500 gpm from the fire protection system
iv. up to 5,400 gpm from the torus

All of these sources can provide makeup water to the SFP in excess of the maximum required 90.77 gpm. The current Fermi 2 SER recognizes that the availability of these makeup water sources satisfies the intent of Regulatory Guide 1.13.

Response to Question 20(c):

Fermi 2 operating procedure ARP 2D1 describes the appropriate operator actions in response to a low water level in the SFP. This procedure directs the operator to "restore spent fuel storage pool level to normal using G4100-F015, FPCC system skimmer surge tanks condensate supply isolation valve". This procedure also identifies sources of emergency makeup water including the RHR system, the RHRSW system and the Fire Protection (FP) system.

Fermi 2 operating procedure ARP 2D13 describes the appropriate operator actions in response to a low water level in the skimmer surge tanks. This procedure directs the operator to "raise water level to normal, slowly throttle open G4100-F015, FPCC system skimmer surge tanks condensate supply isolation valve".

Question 21:

On page 5-16 of Holtec Report HI-992154, it is stated that the fuel rod cladding temperature was evaluated by using a Nusselt-number for laminar flow provided by Robsenow and Hartnett. Please provide a typical Nusselt number and heat transfer coefficient used for the calculation.

Response to Question 21:

Rohsenow and Hartnett ("Handbook of Heat Transfer," 9th Edition, 1973) reports a Nusselt number for fully developed laminar flow over constant heat rate per unit length surfaces (Nu = 4.364). Over small axial segments of fuel cladding, the heat rate is approximately constant, so it is appropriate to use this value. With a hydraulic diameter of 0.540-inches and a thermal conductivity for water of 0.39 Btu/(hr×ft×°F), the heat transfer coefficient is calculated using the equation on page 5-16 as approximately 37.8 Btu/(hr×ft^2×°F). It should be noted that, as stated on page 5-16, an additional crud resistance is included in the overall heat transfer coefficient (U), which is thus reduced to approximately 31.8 Btu/(hr×ft^2×°F).

Table A Batch Decay Heat Loads for Full Core Offload Scenario			
End-of-Cycle Number	Date of Discharge	Non-Dimensional Power Factor	Decay Heat Load (Btu/hr)
1	09/1989	0.01328	203,486.3
2	03/1991	0.01426	218,502.6
3	09/1992	0.01453	222,639.7
4	12/1993	0.01523	233,365.7
5	07/1996	0.01251	191,687.7
6	09/1998	0.01647	252,365.9
7	03/2000	0.01738	266,309.6
8	09/2001	0.01801	275,962.9
9	03/2003	0.01867	286,076.0
10	09/2004	0.01935	296,495.4
11	03/2006	0.02006	307,374.6
12	09/2007	0.01856	284,390.5
13	03/2009	0.01925	294,963.2
14	09/2010	0.01997	305,995.6
15	03/2012	0.02076	318,100.5
16	09/2013	0.02178	333,729.8
17	03/2015	0.02363	362,076.9
18	09/2016	0.02879	441,142.3
19	03/2018	0.04858	744,379.8
20	09/2019	full core @ 160 hrs	36,002,560.1

Table B Batch Decay Heat Loads for Unplanned Full Core Offload Scenario			
End-of-Cycle Number	Date of Discharge	Non-Dimensional Power Factor	Decay Heat Load (Btu/hr)
1	09/1989	0.01344	205,937.9
2	03/1991	0.01444	221,260.7
3	09/1992	0.01470	225,244.6
4	12/1993	0.01542	236,277.0
5	07/1996	0.01266	193,986.2
6	09/1998	0.01666	255,277.2
7	03/2000	0.01759	269,527.4
8	09/2001	0.01823	279,333.9
9	03/2003	0.01890	289,600.2
10	09/2004	0.01959	300,172.9
11	03/2006	0.02030	311,052.1
12	09/2007	0.01879	287,914.7
13	03/2009	0.01948	298,487.4
14	09/2010	0.02022	309,826.2
15	03/2012	0.02106	322,697.4
16	09/2013	0.02224	340,778.2
17	03/2015	0.02474	379,085.1
18	09/2016	0.03253	498,449.4
19	03/2018	0.06846	1,048,996.0
20	03/2019	full core @ 159 hrs	36,095,907.3

Table C Time-to-Boil Coincident with Calculated Peak Bulk Temperatures			
Offload Scenario	Initial Bulk Temperature	Calculated Time-to-Boil	Maximum Boiling Water Loss Rate
Partial Core	125.43°F	13.96 hours	40.07 gpm
Full Core	140.55°F	4.28 hours	89.04 gpm
Unplanned Full Core	141.08°F	4.21 hours	89.99 gpm