Public Service

Wisconsin Public Service Corporation
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August 7, 2000

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

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Proposed Technical Specification Amendment Request PA #167 Spent Fuel Storage – Response to Additional Information

Reference: 1) Letter to Document Control Desk from M.L. Marchi dated November 18, 1999

In reference 1, WPSC submitted proposed amendment (PA) request #167 to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). This PA was submitted to request NRC approval to increase the allowable number of spent fuel assemblies stored in the KNPP spent fuel pool. Since our initial request, discussions among our staffs have taken place discussing the issues associated with this PA. On July 24, 2000, our staffs agreed that WPSC should docket the responses that have been provided to the staff. Therefore, attached please find the KNPP responses to the NRC request for additional information.

Attachment 1 to this letter contains the responses to the Radiological Questions. Attachment 2 contains the responses to Control and Handling of Heavy Loads, Thermal-Hydraulic Considerations and General Issues. Attachment 3 contains the responses to the Structural Issues.

If you should have any questions concerning this proposed amendment, please contact David Molzahn (920) 433-1308 or Jim Brandtjen (920) 388-8421.

Sincerely,

M Marchi

Mark L. Marchi
Vice President-Nuclear

DJM
Attach.
cc - US NRC - Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

A001

ATTACHMENT 1

Letter from M. L. Marchi (WPSC)

to

Document Control Desk (NRC)

Dated

August 7, 2000

Radiological Responses

Requests For Additional Information

Radiological Issues

Question 1

Table 10.1 of the HOLTEC report supporting the license amendment indicates that a dose rate comprised between 2.5 mrem/hr and 5.0 mrem/hr has been assumed by the licensee for estimating dose to personnel during the operations associated with positioning the equipment and installing the new fuel racks. Similarly, the licensee assumed a dose rate comprised between 20 mrem/hr and 35 mrem/hr for operations related to cleaning the transfer canal. Provide justifications for these values. Indicate the main assumptions used to calculate these values.

WPSC Response

The 2.5 mrem/hr to 5.0 mrem/hr range for operations associated with positioning the equipment and installing the new fuel racks is based on current actual radiation levels at the spent fuel pool deck elevation. During rack installation, most work will be performed from the pool deck elevation. Following decontamination of the canal, the canal is expected to have little affect on dose rates at the pool deck elevation.

The dose rate range of 20 mrem/hr to 35 mrem/hr for operations related to cleaning the transfer canal is based on dose data collected for work performed under the radiation work permit associated with the most recent canal decontamination.

Question 2

The license amendment discusses how the increased number of fuel assemblies stored in the Kewaunee spent fuel pool will affect dose rates in adjacent accessible areas to the spent fuel pool. However, it does not specifically address possible radiation zoning changes. State whether the increased storage capacity in the Kewaunee spent fuel pool will necessitate any radiation zoning changes to any of the surrounding areas.

WPSC Response

The calculated dose rates resulting from fuel being stored in the canal storage racks are not expected to require zoning changes in any of the surrounding areas. This is due to the thickness of the canal's concrete walls surrounding the fuel and the age of the fuel to be stored in the canal racks.

There is one area where the predicted dose rate will not be insignificant. This area is at the bottom of the canal next to the south surface of the new partition wall. At this location, the calculated gamma-ray dose rate from the fuel to be stored in the canal is 7.83 mR/hr. However, this area is normally flooded and only after the canal is drained (which is done very infrequently) can an entry into this area be made. Furthermore, special equipment is required to access this area due to the depth of the canal (greater than 40 feet from the canal floor to the pool deck) and thus this area is considered to be inaccessible.

Question 3

Indicate what will the minimum monitoring equipment be for personnel working on the project (TLDs and self-reading dosimeters?). In addition, indicate whether Radiation Protection personnel will constantly monitor the doses to the workers during the spent fuel pool expansion operation.

WPSC Response

TLD badges and electronic dosimeters will be used by personnel working on the project. For the majority of the work, radiation protection personnel will be providing constant coverage, including dose monitoring. A radiation work permit will be developed for this work. Protective equipment, clothing, and monitoring requirements will be identified on the radiation work permit in accordance with good ALARA practices.

Question 4

Indicate if divers are planned to be used for the removal of certain appurtenances in the north or south pools that could result from the installation of the new fuel racks in the fuel transfer canal. If so, what will these divers be equipped with in terms of monitoring equipment?

WPSC Response

The use of divers is not planned for any activity associated with this project.

Question 5

Indicate if a vacuum is planned to be used to remove any radioactive crud, sediment, and other debris in the transfer canal before the new fuel rack modules are installed. If so, discuss your plans to use such a vacuum and indicate how you will dispose of the vacuum filter bags?

WPSC Response

Following decontamination of the canal, it is expected that there will be no significant radiological hazards in the canal. Current plans are to use vacuum cleaners to collect the non-hazardous dust and debris in the canal. All vacuum cleaners will be equipped with HEPA filters. In addition, trained personnel will be used to change the filter bags and monitor the radiation levels of the vacuum cleaners during use. The filters will be disposed of as normal low level radioactive waste. It is expected that the total volume of low level radioactive waste generated due to this project will be less than 50 cubic feet.

Question 6

Discuss how the storage of additional spent fuel assemblies will affect the releases of tritium from the Kewaunee spent fuel pool (the license amendment only addresses Kr-85, Xe-133 and I-131 radionuclides).

WPSC Response

The spent fuel pool contains tritium from two sources. The first source is the tritium from the reactor coolant system. Tritium in the reactor coolant system is a result of neutron capture by ^{10}B (chemical shim) in the reactor coolant. This tritium can only enter the spent fuel pool during refueling outages when the spent fuel pool and the reactor coolant system are interconnected. Since this proposed amendment does not increase the frequency of refueling outages, this source of tritium does not change.

The second source of tritium is a result of neutron capture by ^{10}B in the spent fuel pool water. The decay neutron flux from the old fuel in the spent fuel pool is considerably smaller than the neutron flux in an operating core. Due to the small neutron flux associated with the fuel to be stored in the new racks, the affect on tritium production will be insignificant. Therefore, the release of tritium from the storage of additional spent fuel assemblies in the transfer canal will be insignificant.

Question 7

Discuss how the storage of additional spent fuel assemblies will affect liquid radioactive wastes. Section 10.0, (Radiological Evaluation) of the HOLTEC report only addresses Solid Radwaste and Gaseous Releases.

WPSC Response

As with solid radioactive waste, no significant increase in the quantity of liquid radioactive waste is expected with the expansion of storage capacity. The activity levels of the pool radionuclides will not change since the dominating activity sources are from activation which occurs in the reactor core and from leaking fuel assemblies in storage. The existing fuel assemblies stored in the Kewaunee spent fuel pool have minimal leakage as demonstrated by the current low radioactivity level of the spent fuel pool water. In order to provide confidence that the existing fuel integrity is maintained, the fuel assemblies that will be loaded into the new racks will use procedures that include the same handling methods as those previously approved. Due to the proven method of fuel movement and the integrity of the fuel to be stored in the new racks, no significant increase in pool radioactivity associated with this project is anticipated. As the proposed amendment negligibly affects the quantity of radioactivity in the spent fuel pool system, it is concluded that the storage of additional spent fuel assemblies will have a negligible impact on liquid radwaste.

Question 8

Indicate if the frequency of resin changeout is expected to increase during the installation of the new fuel racks.

WPSC Response

The resin in the spent fuel pool demineralizer is typically replaced every 12 to 15 months. During the loading of the fuel into the new racks, it is possible that fuel movement may stir up a small amount of settled contamination. However, it is expected this will have an insignificant affect on the frequency of resin change out.

ATTACHMENT 2

Letter from M. L. Marchi (WPSC)

to

Document Control Desk (NRC)

Dated

August 7, 2000

Control and Handling of Heavy Loads

Thermal-Hydraulic Considerations

General Issues

Requests For Additional Information

Control and Handling of Heavy Loads

Question 1

Section 11.0, Installation (Attachment 5, letter from Marchi to NRC, dated 11/18/99), states that the new racks will not be carried over any regions in the pools containing fuel. NUREG-0612, Section 5.1.2(3)(b), specifies the need for mechanical stops or electrical interlocks which prevent movement of the overhead crane load block over or within 25 feet horizontal of the hot spent fuel. Will there be any hot fuel in the spent fuel pool during the proposed rack placement? If so, please verify that the storage of the spent fuel will be controlled such that this requirement will be satisfied during the installation of the proposed new racks.

WPSC Response

The KNPP Technical Specifications (TS 3.8.a.8) already prohibits the movement of a heavy load over either spent fuel pool. Although the technical specifications currently allow the placement of additional fuel storage racks, the new racks are not permitted to traverse directly above spent fuel stored in the pools. Electrical interlocks have already been installed to prevent the crane from inadvertently traveling over either of the pools. The route that the new spent fuel canal racks will follow does not traverse over either of the existing spent fuel pools, and as required by KNPP administrative controls, must be approved by the Plant Operations Review Committee prior to use. The canal area to be racked currently does not contain any spent fuel and will be isolated from the existing pools during rack installation.

Question 2

Attachment 1 (letter from Marchi to NRC, dated 11/18/99) concludes that the probability of an accidental fuel assembly drop is primarily influenced by the methods used to lift the fuel. The increased frequency of fuel handling resulting from this proposed reracking appears to influence the probability as well. What effect will the increased fuel handling due to the shuffling of the fuel assemblies into the new racks, and due to the increased capacity of the pool, have on this conclusion?

WPSC Response

The incremental amount of fuel movement required to load the canal racks is insignificant when compared to the fuel movement that has occurred since 1974 and the fuel movement planned through 2013. The fuel that will be loaded into the canal racks will remain there until it is ultimately removed from the pool. Therefore, when considering the amount of fuel movement required to operate KNPP for 40 years, the movement required to install the 215 fuel assemblies in the canal racks is considered to have an insignificant effect on the probability of an accidental fuel assembly drop.

Question 3

The proposed revision to Technical Specification 5.4.c specifies restrictions on fuel placement in the pool. Attachment 1 (letter from Marchi to NRC, dated 11/18/99) states that the mispositioning event for the new canal racks represents a change from the previously analyzed condition, since Kewaunee

currently has no restrictions on fuel placement. It is further stated that the mispositioning event in the canal does not represent a new or different kind of accident. What is the basis of this statement, since there were no placement restrictions associated with the spent fuel pool previously?

WPSC Response

The mispositioning event for the new canal racks does represent a change from the previously analyzed condition, since Kewaunee currently has no restrictions on fuel placement in the existing racks. However, the analyzed mispositioning event that could result today, is due to a fuel assembly being positioned outside one of the existing racks, between the rack and the spent fuel pool wall. The mispositioning event of a fuel assembly outside the existing rack was previously evaluated and determined to be acceptable.

The mispositioning of a fuel assembly in the new racks results from the placement of a fresh (unburned) fuel assembly in the canal rack. The results of this mispositioning event were determined to be acceptable using similar acceptance criteria to those used for the previous mispositioning event. Therefore, since both events represent the mispositioning of a fuel assembly, the mispositioning of an assembly in the canal racks is not considered to represent a new or different kind of accident.

Question 4

Attachment 1 (letter from Marchi to NRC, dated 11/18/99) states that an accidental drop of the rack during installation is not a new or different type of event since it is bounded by the previous reracking project which used significantly larger rack modules. What is the weight of the two different rack module sizes (5 x 10 and 5 x 11) and the lift rig? How does this weight compare to the weights of the previously installed racks in the north and south pools?

WPSC Response

The nominal weight of the canal rack modules is 6,810 lbs. and 6,130 lbs. for the (5x11) and (5x10) modules respectively. The rack modules in the existing pools are 9x10 and weigh approximately 42,000 lbs. The weights of the lifting rigs are small in comparison with the weights of the rack modules. Therefore, the weight associated with a canal rack module with the associated lifting rig is significantly less than that for the existing racks.

Question 5

Table 11.2.1 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) indicates that the crane operators will be trained and qualified. Section 5.1.1(3) of NUREG-0612 specifies that crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976. Please verify that the training will be in accordance with this requirement. In addition, Section 9.0 (Attachment 2, letter from Marchi to NRC, dated 11/18/99) states that an additional person will be present during rack movement. What are the responsibilities of this person, and what training does this person receive?

WPSC Response

As required by KNPP administrative procedures, only crane operators that have been trained and qualified in accordance with the KNPP crane-training program which is in accordance with Chapter 2-3 of ANSI B30.2-1976 will be used.

The additional person that will be present during rack movement is the signalman who provides the hand signals that may be required during load movement. This individual may also provide additional guidance to the crane operator and will have completed KNPP annual Control of Heavy Loads refresher training.

Question 6

Table 11.2.1 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) indicates that the cranes will be inspected and tested prior to use in the rerack. Section 9.0, Operational Issues (Attachment 2), states that the crane inspection will be performed per KNPP General Nuclear Procedure GNP 8.12.1. Section 5.1.1(6) of NUREG-0612 specifies that this inspection should be performed in accordance with Chapter 2-2 of ANSI B30.2-1976. Please verify that the inspections will be in accordance with this requirement.

WPSC Response

The auxiliary building crane is tested, maintained, and inspected in a manner that satisfies Chapter 2-2 of ANSI B30.2-1976. The KNPP program was found to be consistent with the guidelines of NUREG-0612 in a Safety Evaluation Report to C.W. Giesler from S.A. Varga dated March 16, 1984.

Question 7

The proposed placement of the new spent fuel racks in the transfer canal extends storage to areas which may not have originally been considered for storage. Please verify that the fuel handling system will have sufficient travel to access all cells of the new spent fuel racks.

WPSC Response

The KNPP fuel handling system will have sufficient capability to access the cells of the new spent fuel racks.

Question 8

Section 11.7 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) states that the racks will be removed from the trailer using a "suitably rated crane," and "using two independent overhead hooks, or a single overhead hook" uprighted into the vertical position. It is not clear if the auxiliary building crane will be used for these operations, or some other plant crane. Please specify which crane(s) will be used for these operations (including the raising of the racks to the refuel floor elevation), and if different from the auxiliary building crane, describe what inspections and precautions will be used?

WPSC Response

The racks will be removed from the truck, in the horizontal position, using two chain hoists and a spreader beam hung from the fuel-handling crane (also known as the auxiliary building crane). Each hoist will be rated for at least 65% of the total lifted weight to account for unequal loading and dynamic load factors. The rack upending will be performed using an upending frame (L-frame) and the chain hoists and spreader beam. The fuel-handling crane will be used to lift the upending frame off the ground, and the chain hoists will be used to rotate the upending frame and rack to the near vertical position. The racks will be raised to the refueling floor using a NUREG-0612 lift rig specifically designed for the racks and the fuel-handling crane. The heaviest rack weighs less than 3.5 tons and the lift rig weighs less than 1/2-ton. The main hook of the fuel-handling crane is rated for 125 tons.

Question 9

The design of the SFP in Kewaunee appears to allow spent fuel to be transferred to either the north or south pools through specific transfer gates (one for each pool) from the fuel transfer canals. With the proposed conversion of the north transfer canal to a new storage area, will it be necessary to modify procedures for the future fuel transfer? If so, what are your plans?

WPSC Response

The proposed conversion will not require a modification of the fuel handling procedures. The fuel handling procedures do not describe the path of movement to the destination cell location.

Thermal-Hydraulic Considerations

Question 10

Holtec stated on Page 5-8 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) that the differential equation on Page 5-6 was solved numerically to obtain the transient thermal response of the spent fuel pool. To solve the equation, it is necessary to know either p (heat exchanger temperature effectiveness), or t_o (coolant outlet temperature) for the Q_{HX} term on Page 5-7. Please explain how these values are obtained.

WPSC Response

As stated on page 5-7, the temperature effectiveness is calculated using the following equation:

$$p = \frac{t_o - t_i}{T - t_i}$$

where t_o is the coolant outlet temperature (°F), t_i is the coolant water inlet temperature (°F) and T is the SFPs bulk water temperature (°F). These heat exchanger inlet and outlet temperatures are referred to as “terminal temperatures”. Version 5.04 of the Holtec QA validated computer program STER was used to model the thermal performance using the SFP heat exchanger geometric data obtained from manufacturer data sheets and drawings. Four sets of terminal temperatures were determined for four different values of the SFP bulk water temperature using the STER computer program, which yielded the temperature effectiveness as a function of SFP bulk temperature. Design-

basis maximum fouling levels and a 10% tube plugging allowance were included in the STER model to ensure conservative results.

Question 11

To solve the above-mentioned differential equation, information on the heat exchanger performance is needed. Please provide the following additional data or explain why they are not needed:

a. The heat transfer rate of the SFPCS heat exchangers is provided at a fixed shell side temperature (66°F) and tube side temperature (120°F) on Page 5-2. However, the temperatures of the shell side cooling water from service water system and the tube side water from the SFP may be higher than these temperatures (e.g., 80°F is assumed for the shell side cooling water in Table 5.4.1). Please provide the heat transfer rates as a function of the tube side SFP temperature at the expected maximum temperature of the shell side cooling water. Please also justify the expected maximum temperature of the cooling water.

b. With regard to the service water temperature for the SFPCS and RHR heat exchangers, Holtec indicates that:
   (i) The shell side water (service water) temperature of the SFPCS heat exchangers is 66°F (in Page 5-2).
   (ii) SFPCS heat exchanger coolant (service water) inlet temperature is 80°F (Table 5.4.1).
   Please clarify the above discrepancy. Also, in Page 5-5, Holtec states that in all scenarios, the service water that removes heat from the SFPCS and CCW heat exchangers is assumed to be at its design maximum temperature. Please clarify what is the design maximum temperature for these heat exchangers (66°F, 80°F, or some higher temperatures).

c. The above heat transfer rate is provided at the tube side water flow rate of 425,000 lb./hr. (i.e., about 850 gpm), while the SFPCS pump capacity is given at 600 gpm for each and 1080 gpm for both. If different flow rates than 850 gpm are expected in the actual operation, please provide the above functions at these actual flow rates (which accounts for the maximum surface fouling resistance and tube plugging).

d. The heat transfer rate of the RHR heat exchanger is provided at the tube side water flow rate of 1,000,000 lb./hr. (i.e., 2000 gpm), while 280 gpm (140,000 lb./hr.) is assumed in Table 5.4.1. What is the actual flow rate to the "A" RHR heat exchanger when it is in the SFP cooling mode based on the piping configuration between the SFP and the RHR? Is the RHR pump running and aligned to the RHR heat exchanger when the RHR heat exchanger is in the spent fuel pool cooling mode? Please provide the RHR heat transfer rates as a function of the tube side SFP temperature at the expected flow (which accounts for the maximum surface fouling resistance and tube plugging).

WPSC Response

The SFPCS and RHR heat exchanger performance values presented on pages 5-2 and 5-3 are provided as part of a general system description. These values correspond to the nominal design performance of the two cooling systems. The performance values in Subsection 5.2 are not intended to represent the limiting performance of the respective systems in the alignments used during

refueling outages, but rather to provide general background information of the type usually provided in an FSAR. Actual heat exchanger performance is dependent on system conditions (i.e., flow rates, heat sink temperatures, etc). During a refueling outage, when SFP decay heat loads and temperatures are higher and system alignments are different, the SFPCS and RHR heat exchanger performance values will vary from the nominal values. The values listed in Table 5.4.1 are the bounding design values for the refueling outage conditions and are utilized in performing the thermal-hydraulic evaluations.

a. As stated above, the performance values on page 5-2 are provided as part of a general system description, and do not represent the limiting refueling outage performance. The following table presents the SFPCS heat exchanger heat transfer rates, as functions of temperature, for the limiting refueling outage conditions (i.e., for the conditions listed in Table 5.4.1). As indicated in Table 5.4.1, the SFPCS temperature (Service Water temperature) is assumed to be 80°F.

Number of Operating SFPCS Pumps	SFP Flow Rate to SFPCS HX (gpm)	SFP Bulk Temperature to SFPCS HX (°F)	SFPCS HX Heat Removal Rate (Btu/hr)
1	600	100	3.10×10^6
		125	7.05×10^6
		150	11.06×10^6
		175	15.09×10^6
2	1080	100	3.86×10^6
		125	8.78×10^6
		150	13.77×10^6
		175	18.82×10^6
2	800	100	3.49×10^6
		125	7.93×10^6
		150	12.45×10^6
		175	17.02×10^6

There are three sets of performance data in the above table because heat exchanger performance is highly dependent on the water flow rate. The SFP water flow rate to the SFPCS heat exchanger varies with the number of pumps operating (i.e., single failure or no single failure) and the number of heat exchangers being supplied (i.e., SFPCS only or SFPCS and RHR heat exchanger in parallel).

b. As stated above, the performance values on page 5-2 are provided as part of a general system description, and do not represent the limiting refueling outage performance. The service water (SW) and component cooling water (CCW) temperatures listed in Table 5.4.1 are 80°F and 88°F, respectively. The 80°F SW temperature is the design-basis maximum temperature for the SW system. The 88°F CCW temperature is the maximum expected CCW temperature at the inlet of the RHR heat exchanger 132 hours after reactor shut down for a SW temperature of 80°F. The CCW temperature was determined using the KNPP validated Proto-hx software and is conservative. The conservatism is due to using the higher core

decay heat load associated with 132 hours after shutdown rather than the heat load for 148 hours after shutdown which is the earliest time at which fuel transfer could begin.

c. As stated above, the performance values on page 5-2 are provided as part of a general system description, and were not used in performing the analyses. The table in the response to Question 11a, above, provides the SFPCS heat exchanger performance as a function of both SFP bulk temperature and SFP water flow rate. The service water flow rate to the SFPCS heat exchanger, given in Table 5.4.1 as 860 gpm, is the same for all system lineups.

Recent flow test data for the SFP heat exchanger in conjunction with computer analyses have demonstrated that the SFP cooling system would be capable of the assumed flow rates under the conditions of 10 percent tube plugging, long term fouling conditions, and reductions to account for instrument accuracy.

d. As stated above, the performance values on page 5-2 are provided as part of a general system description, and do not represent the limiting refueling outage performance. The performance values presented in Table 5.4.1, which were used for analysis, are the design maximum values for the alignments used during refueling outages. For operation in the SFP cooling mode the flow rate to the RHR heat exchanger was assumed to be 280 gpm. The 280 gpm flow rate through the RHR heat exchanger was arrived at by using a computer analysis that modeled the system piping configuration along with existing plant data. The analysis accounted for tube plugging and long term fouling. The 280 gpm flow to the RHR heat exchanger is only provided by the SFPCS pumps. The RHR pumps are not used for SFP cooling. The following table presents the RHR heat exchanger heat transfer rate, as a function of temperature, for the SFP cooling mode conditions (i.e., for the conditions listed in Table 5.4.1).

SFP Bulk Temperature to RHR HX (°F)	RHR HX Heat Removal Rate (Btu/hr)
100	1.35×10^6
125	4.19×10^6
150	7.05×10^6
175	9.93×10^6

Question 12

It is stated on Page 5-8 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) that the decay heat load was calculated using the Holtec LONGOR Program, while Page 5-2 (Attachment 2, letter from Marchi to NRC, dated 11/18/99) states that it is calculated in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." Please clarify the above discrepancy.

WPSC Response

The decay heat loads were calculated using the Holtec LONGOR program. This program has previously been reviewed and accepted by the NRC for calculating decay heat loads in SFP capacity expansion applications.

Question 13

It is stated on Pages 5-2 and 5-4 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) that the "A" RHR will be aligned to the SFP if the service water temperature exceeds 60°F. Please describe the administrative control procedures or Technical Specifications which ensure the RHR alignment. Please also discuss the criteria of the set-point (at the service water temperature of 60°F) required for the "A" RHR to be aligned to cool the SFP.

WPSC Response

To ensure the RHR alignment, the following requirements will be added to the Reactor Engineering procedure that controls fuel movement during refueling outages. Prior to removing the last fuel assembly from the reactor vessel during a full core off-load, the water temperature within the main Service Water piping will be determined using existing instrumentation. If the SW temperature is ≥ 60F, the "A" RHR hx will be aligned as described in section 5.8.1. The SW temperature of 60°F was chosen because the SFP hx alone was shown to have adequate capacity at a SW temperature of 60°F or lower under bounding conditions.

NOTE:
The intent of this note is to inform you that a project of this nature has an impact on a number of different plant groups. Specifically, the plant operations group has expressed it's desire to continue to use the SFP cooling system in the typical SFP cooling alignment under conditions where the heat load is within the heat removal capacity of the SFP hx. The KNPP believes that alignment of the 1A RHR hx should only be performed when additional cooling is needed. Thus additional evaluations may be performed to support this goal. The 60°F SW temperature action limit is based on bounding conditions. Prior to an actual full core off-load, the conditions specific to the outage may be reviewed to evaluate increasing this action limit if justified. Conditions that differ from the bounding conditions used in the analyses such as the number of plugged tubes in the SFP hx, increasing the in-core hold time, etc. may be considered.

Also, to facilitate implementation of the RHR hx alignment, the timing of the alignment may be based on the equivalent time after fuel transfer begins, rather than being based on the time after the last assembly is removed from the reactor vessel. The equivalent time would be the assumed maximum fuel transfer rate multiplied by the number of fuel assemblies, plus the specified allowable time for alignment. For the normal full core off-load case this would be (121 assemblies in core) ÷ (6 fuel assemblies/hr.) + 1.8 hours = 22 hours.

Any change concerning the SW action limit and RHR hx alignment timing would be evaluated in accordance with 10CFR 50.59, performed in accordance with KNPP procedure control requirements, and approved by the KNPP Plant Operations Review Committee (PORC) prior to implementation.

Question 14

The RHR system is not available to cool the SFP during the fuel transfer for the full core offload, since it will be aligned to the SFP cooling only after the completion of the fuel transfer. Is this delay (estimated to be about 20 hours at the transfer rate of six assemblies per hour) accounted for in

estimating the maximum bulk temperature in Table 5.8.1 (Attachment 5, letter from Marchi to NRC dated 11/18/99) for the full core transfer scenarios (Scenarios 2 & 3)?

WPSC Response

The analysis accounted for the time period when the RHR heat exchanger is not providing cooling to the SFP. As stated on page 5-2 with respect to full-core offloads, "following the completion of fuel transfer, the heat removal capacity of the SFPCS may be increased by aligning a Residual Heat Removal (RHR) System heat exchanger in parallel with the SFPCS heat exchanger." As stated on page 5-16, "the parallel alignment of the RHR heat exchanger is performed after the end of fuel transfer and can be accomplished in less than one hour. For either emergency or planned full-core discharges, with a 148 hour required in-core hold time and a 20.2 hour transfer time, the end of fuel transfer occurs 168.2 hours after reactor shutdown. This allows 1.8 and 2.8 hours for parallel alignment of the RHR heat exchanger for the emergency and planned full-core discharge scenarios, respectively, before the 150°F temperature limit would be exceeded." The results presented in Table 5.8.1 do, therefore, correctly reflect the fact that the RHR system is not available for SFP cooling until the completion of the core offload.

Question 15

Please explain the rationale for the number of the assemblies specified for Scenario 2, Emergency Full Core Discharge, in the table on Page 5-3; specifically, why does the table indicate 25 assemblies, as opposed to 48, in addition to the full core?

WPSC Response

As stated on page 5-4 with respect to the emergency full core discharge of 121 assemblies, "the 121 discharged assemblies are separated into three distinct groups: 48 assemblies with 3 years plus 30 days of irradiation at full power, 48 assemblies with 1.5 years plus 30 days of irradiation at full power and 25 assemblies with 30 days of irradiation at full power." As stated in page 5-6 with respect to the emergency full core discharge, "the assemblies in the core are split into three regions with burnup levels corresponding to 30 days at power, once-burned plus 30 days at power and twice-burned plus 30 days at power. The twice-burned plus 30 days and once-burned plus 30 days regions are each assumed to be the size of the maximum refueling batch size, resulting in the maximum number of assemblies having the highest possible burnups." This distribution of exposure conservatively maximizes the decay heat generation of the most recently discharged fuel assemblies (the assemblies transferred to the SFP during the emergency full core discharge). In order to maintain consistency between the emergency discharge and the assemblies discharged in the preceding planned refueling (30 days earlier), the number of assemblies discharged in the planned refueling had to be equal to the number of assemblies with only 30 days of exposure. Thus, the previous discharged fuel batch size was set equal to 25 assemblies.

Question 16

What pool temperature is assumed at the time of loss of cooling to calculate the time-to-boil in Table 5.5.1 (Attachment 5, letter from Marchi to NRC, dated 11/18/99)?

WPSC Response

As stated on page 5-9, "the loss of forced cooling is assumed to occur coincident with the peak SFPs bulk temperature." The initial bulk water temperature for each evaluated time-to-boil scenario is, therefore, the peak bulk temperature listed in Table 5.8.1 for the corresponding discharge scenario.

Question 17

Tables 5.8.1 and 5.8.2 (Attachment 5, letter from Marchi to NRC, dated 11/18/99) indicate that the "Emergency Full-Core Discharge" case (Scenario 2) has a lower bulk temperature, lower net heat load, a lower boil off rate, and a longer time-to-boil than "Planned-Full Core Discharge" (Scenario 3). This appears to be contrary to the expectation since Scenario 2 is generally expected to be the worst case (which is why this scenario is considered in addition to the normal full-core offload scenario). Please explain.

WPSC Response

The bulk temperature and the time-to-boil are directly dependent on the decay heat load, so only the perceived decay heat load discrepancy needs to be addressed. Decay heat generation of spent fuel is, to a large degree, dependent primarily on the fuel burnup. Thus, an examination of the differences in fuel burnup is necessary.

The expectation that the emergency full-core discharge will have a higher decay heat load than the planned full-core discharge is based on the impossible assumption that the fuel assemblies in the reactor will accumulate an entire cycle of exposure in only 30 days. If this condition were true, the reduced cooling time of the previously discharged refueling batch in the emergency discharge scenario would result in the total decay heat load being higher. This condition, however, is not true.

It should be noted that 30 days is less than 6% of the normal 18-month cycle length. It is not possible for 18 months of burnup to be accumulated in just 30 days. If this fact is recognized, it becomes apparent that while the shorter cooling time of the previously discharged refueling batch does increase its decay heat load contribution, the reduced burnup of the discharged full core 30 days later results in a reduction of its decay heat contribution. As the reduction in decay heat from the discharged core exceeds the increase in decay heat from the previously discharged refueling batch, the net result is a decrease in the total decay heat load. This decrease in the total decay heat load for the emergency discharge scenario results in the observed decrease in bulk temperature and increase in time-to-boil.

Question 18

The new storage section in the north transfer canal is connected to the main pool through a relatively narrow slot. This new storage area does not have direct cooling. Although this new storage area is cooled by natural convection through this slot (Page 5-18, Attachment 5), it is not obvious that this cooling is adequate. Please provide the temperature and velocity distributions in the new storage section through the middle plane in the north-south direction (similar to that shown on Figure 5.8.2). Explain the mechanism how this section is cooled using the above temperature and velocity profiles. What is the estimated maximum temperature difference between this section and the pool bulk temperature? What are the dimensions (width and height) of the transfer slot? Are there any temperature monitors in the new storage section?

WPSC Response

Figures 1, 2 and 3 present the velocity vectors in a vertical plane through the middle of the transfer canal racks for the partial core, full core and emergency full core discharges, respectively. Figures 4 through 6 present the corresponding temperature contours in the same plane.

Although not identical, the velocity vectors in Figures 1 through 3 all show a similar pattern; water from both ends of the new storage section flows toward the slot that connects the new storage section to the north spent fuel pool. The general pattern is that the heated water issuing from the top of the racks is swept horizontally across the top of the racks toward the north and south ends of the new storage section, where it rises toward the water surface and flows to the slot. This demonstrates that there is a vigorous, passive mechanism for moving heated water from the fuel assemblies in the new storage racks through the slot and into the north spent fuel pool, where it can be subsequently transferred to the SFPCS.

The temperature contours in Figures 4 through 6 demonstrate the efficiency of the above described water flow pattern on removing heat from the stored spent fuel assemblies. The contours indicate that there is little thermal stratification in the water of the new storage area, with temperature variations of approximately 5°F in all three scenarios. All scenarios have similar heat loads in the new storage area, so the similarity of the temperature gradients is not unexpected. The uniformity of the temperatures in the new storage area demonstrates that the heat generated in the stored fuel assemblies is not permitted to build-up large temperature gradients in the water, which is indicative of a vigorous exchange of water with the north spent fuel pool.

Figures 4 through 6 can also be examined to determine the difference between the new storage area temperatures and the pool bulk temperatures. The following table presents a comparison of the new storage area temperatures with the corresponding bulk pool temperatures.

Discharge Scenario	Estimated New Storage Area Bulk Temperature (°F)	Spent Fuel Pool Bulk Temperature (°F)
Partial Core	139	140
Full Core	146	150
Emergency Full Core	149	150

An examination of the values in this table show that there is practically no difference between the temperatures in the new storage area and the corresponding bulk pool temperatures. This is not surprising, as the majority of the total discharged fuel decay heat is located in the existing spent fuel pools and not in the new storage area. The relatively small decay heat generation in the transfer canal ensures that a small temperature difference between the new storage area and the north pool generates sufficient buoyancy forces to transfer the heat from the new storage area to the north pool.

The slot that connects the new storage area to the north spent fuel pool is 2'-6" wide by 22'-11 1/4" high (to the as-modeled water depth). To simplify the geometric modeling of the slot, a slightly smaller height is used in the analysis model, with the bottom sill of the slot raised by 7 inches. This large opening (over 57 ft^2, modeled as over 55 ft^2) provides a low resistance flow path between the

new storage area and the north spent fuel pool. There will be no temperature monitors in the new storage area.

Question 19

Does the "SFP Net Water Volume" specified in Table 5.5.1 include the water in the transfer canal? Or is it the volume when the transfer canal slot is closed? Please explain which is more conservative.

WPSC Response

The SFP net water volume used in all thermal transient evaluations, including the time-to-boil calculations, includes the water in the existing spent fuel pools and the water in the new pool created in the north end of the transfer canal (i.e., north of the new partition wall). As stated on page 1-2, "the north pool transfer slot, which connects the north half of the transfer canal and the north pool, will be permanently opened," so it is appropriate to include this body of water in the total. The water in the south portion of the transfer canal (i.e., south of the new partition wall) is not included.

Question 20

The north canal transfer slot (between the north pool and the new storage section) should be permanently opened not to isolate the new storage section. Please describe procedures to ensure this slot is never closed (currently, there is a gate in this slot to facilitate draining of the transfer canal).

WPSC Response

Prior to the placement of fuel assemblies in the canal racks, permanently mounted blocks will be installed at the top of the gate guides for the north gate opening. This will physically prevent the inadvertent installation of a gate which would isolate the new storage area.

General Issues

Question 21

On Page 5-15 (Attachment 5, letter from Marchi to NRC, dated 11/18/99), it is stated that the fuel rod cladding temperature was evaluated by using a Nusselt–number for laminar flow provided by Rohsenow and Hartnett. Please provide a typical Nusselt-number and heat transfer coefficient used for the calculation. (Reactor Systems Branch)

WPSC Response

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 0.636-inches and a thermal conductivity for water of 0.377 Btu/(hr×ft×°F), the heat transfer coefficient is calculated using the equation on page 5-15 as approximately 31 Btu/(hr×ft^2×°F). It should be noted that, as stated on page 5-15, an additional crud resistance is included in the overall heat transfer coefficient (U), which is thus reduced to approximately 30.6 Btu/(hr×ft^2×°F).

Question 22

As stated in the description of site operations (Section 11.3g of Attachment 5, letter from Marchi to NRC, dated 11/18/99), the installation of the proposed new racks will be performed dry. Will all of the rack installation operations be performed from the refuel floor (e.g., installation of rack bearing pads, removal of any interfering protrusions on the floor of the transfer canal floor), or will it be necessary for personnel to go down into the transfer canal? If so, what effect will these operations have on the personnel exposure estimates provided in Section 10, Radiological Evaluation? (Radiation Protection)

WPSC Response

The operation which is expected to cause the most personnel exposure is the cleaning/ decontaminating of the canal following canal dewatering. After the canal has been decontaminated, the radiation level in the canal will be significantly reduced. The majority of the rack installation work will be performed from the refueling floor elevation. The amount of work associated with removing interferences is small and this work will be performed at or near the refueling floor elevation. Some work from within the canal will be necessary but the number of personnel working in the canal and the time required for work in the canal will be small. All anticipated work associated with rack installation, including work from within the canal, was considered in the development of Table 10.1. The dose rates and required duration for the tasks that were used are considered to provide a conservative estimate for total personnel exposure associated with rack installation.

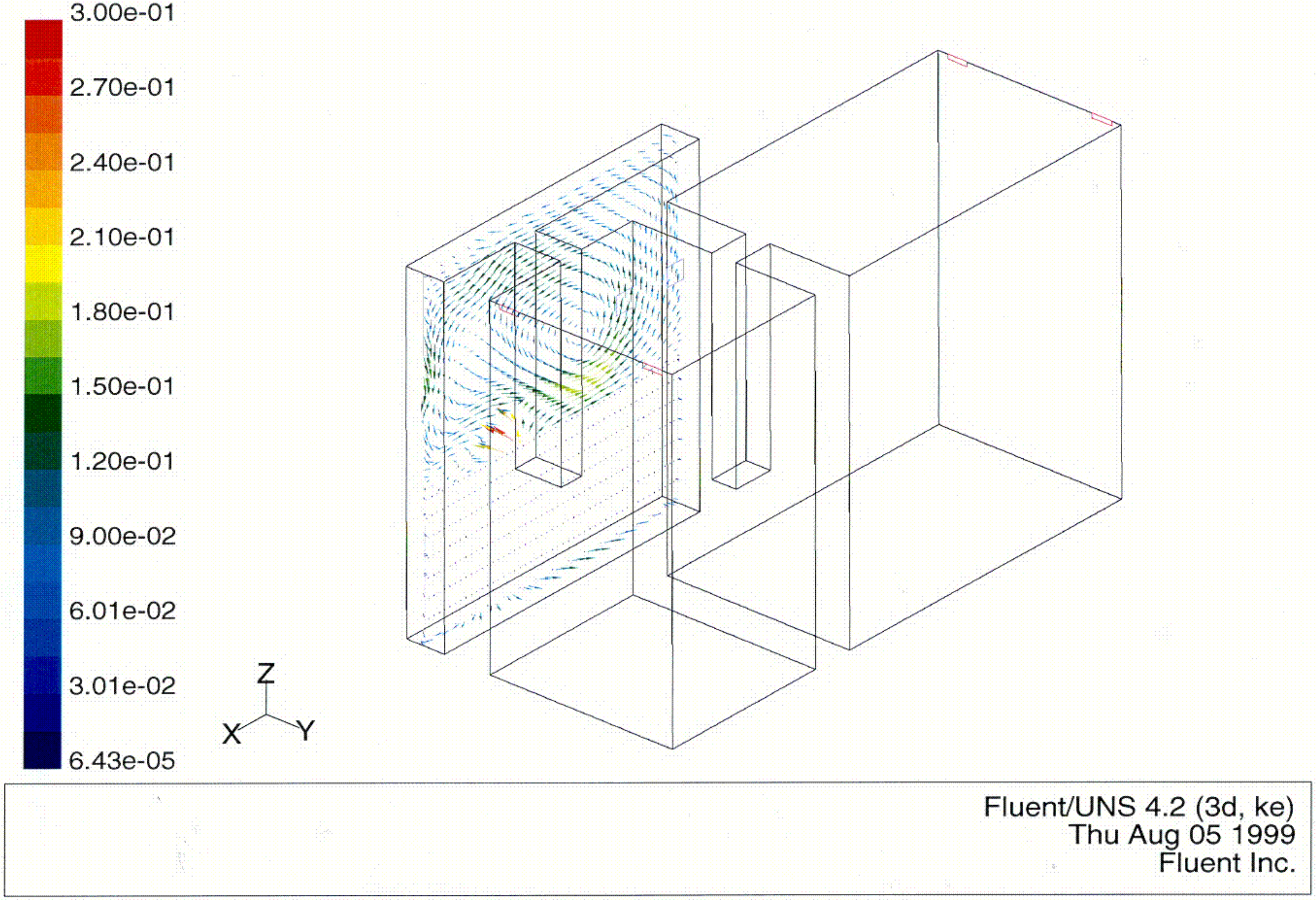
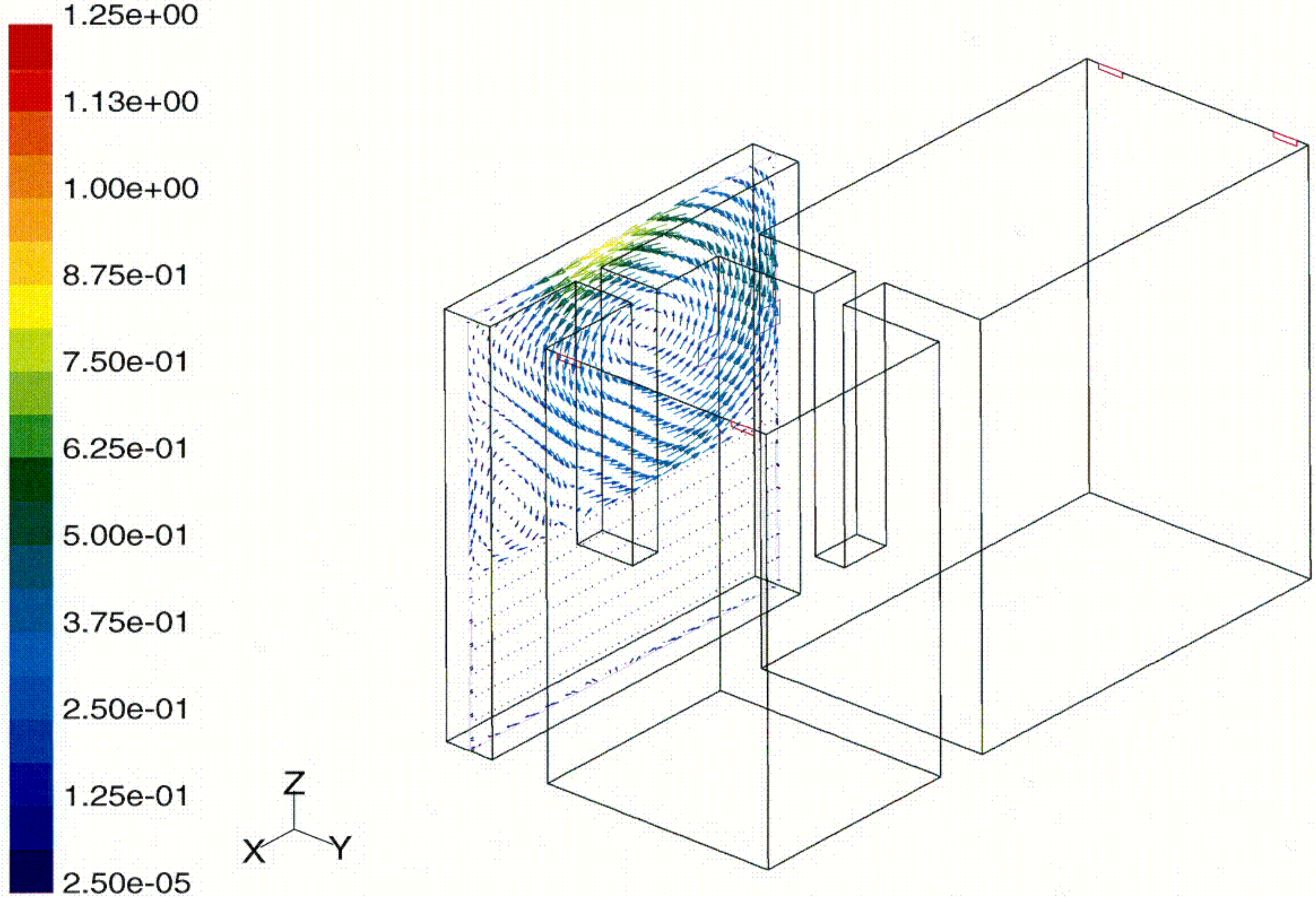


Figure 1: Velocity vectors (ft/s) in a plane through the center of the Kewaunee transfer canal racks (Partial Core Discharge Scenario)

C1



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Thu Aug 05 1999
Fluent Inc.

Figure 2: Velocity vectors (ft/s) in a plane through the center of the Kewaunee transfer canal racks (Full Core Discharge Scenario)

C2

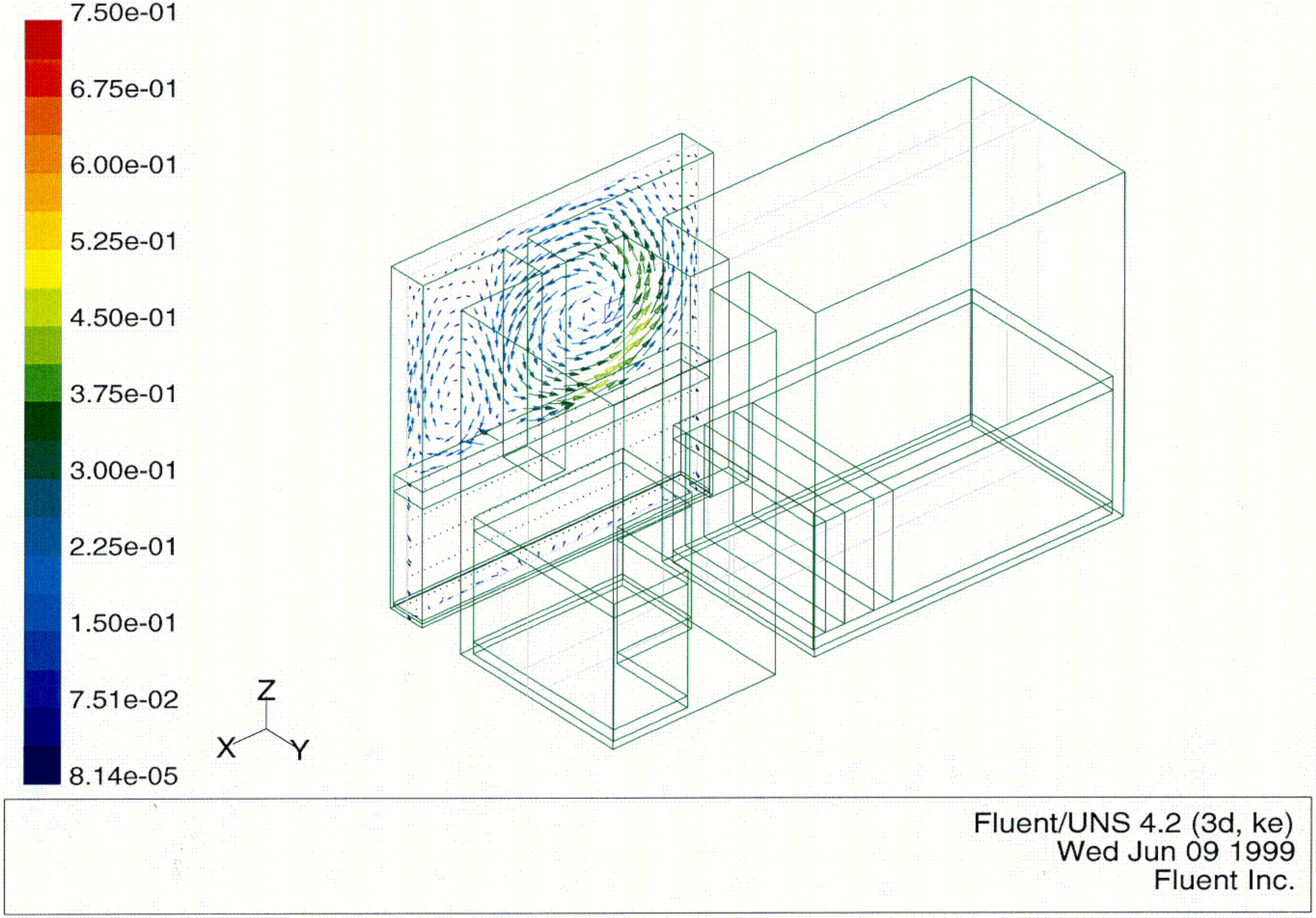

Figure 3: Velocity vectors (ft/s) in a plane through the center of the Kewaunee transfer canal racks (Emergency Full Core Discharge Scenario)

C3

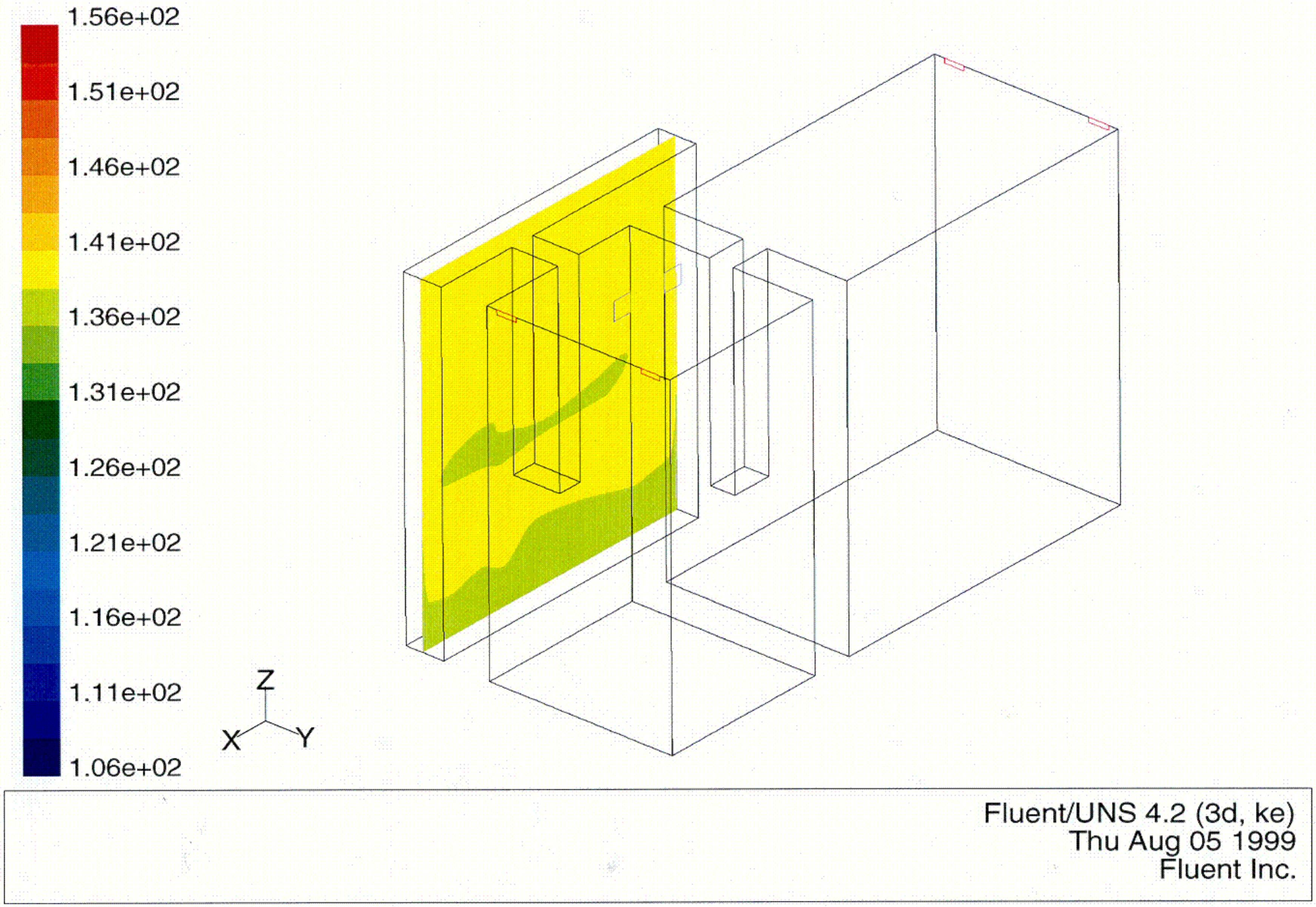

Figure 4: Temperature (degree F) contours in a plane through the middle of the Kewaunee transfer canal (Partial Core Discharge Scenario)

C4

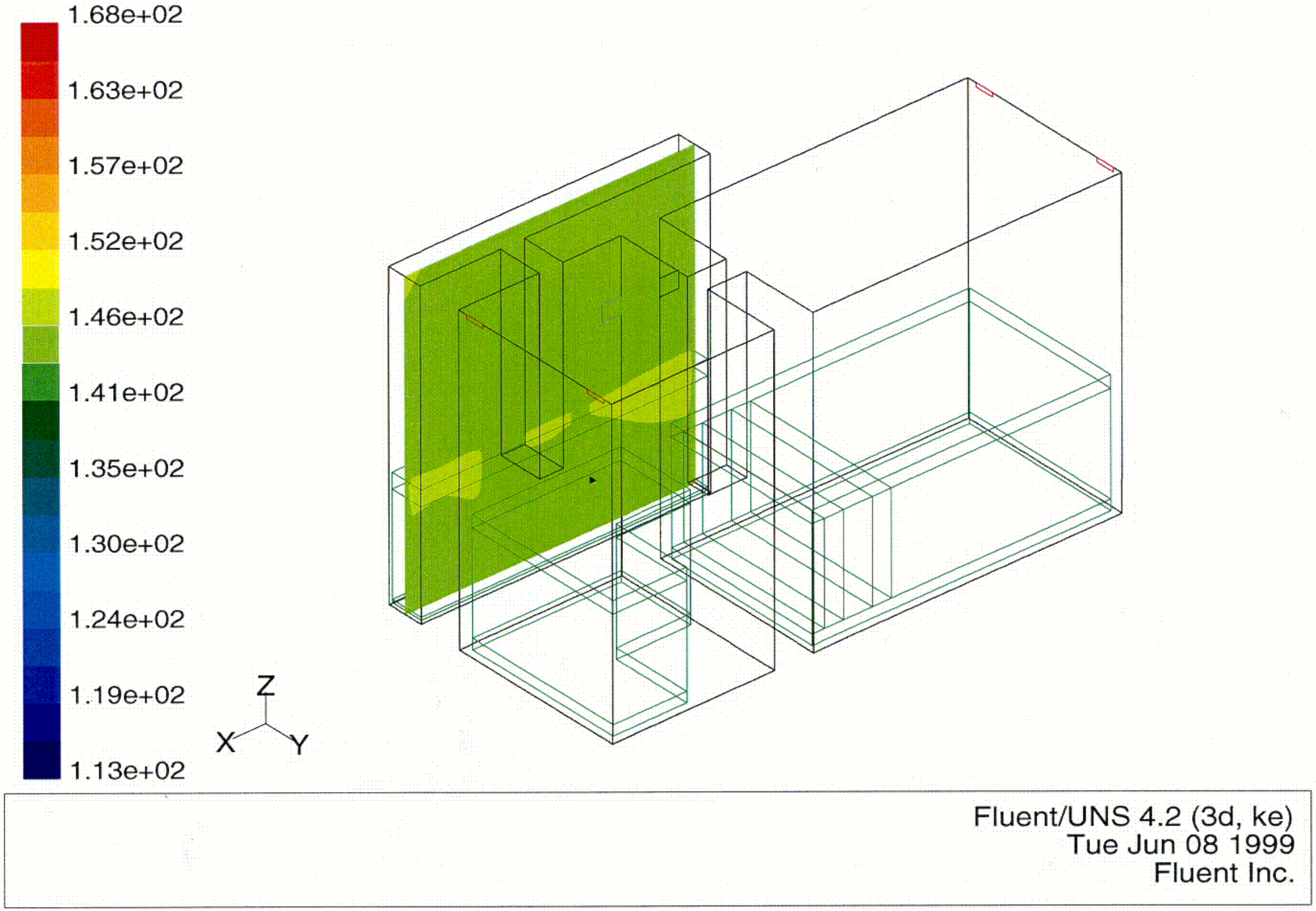

Figure 5: Temperature (degree F) contours in a plane through the middle of the Kewaunee transfer canal racks (Full Core Discharge Scenario)

C5

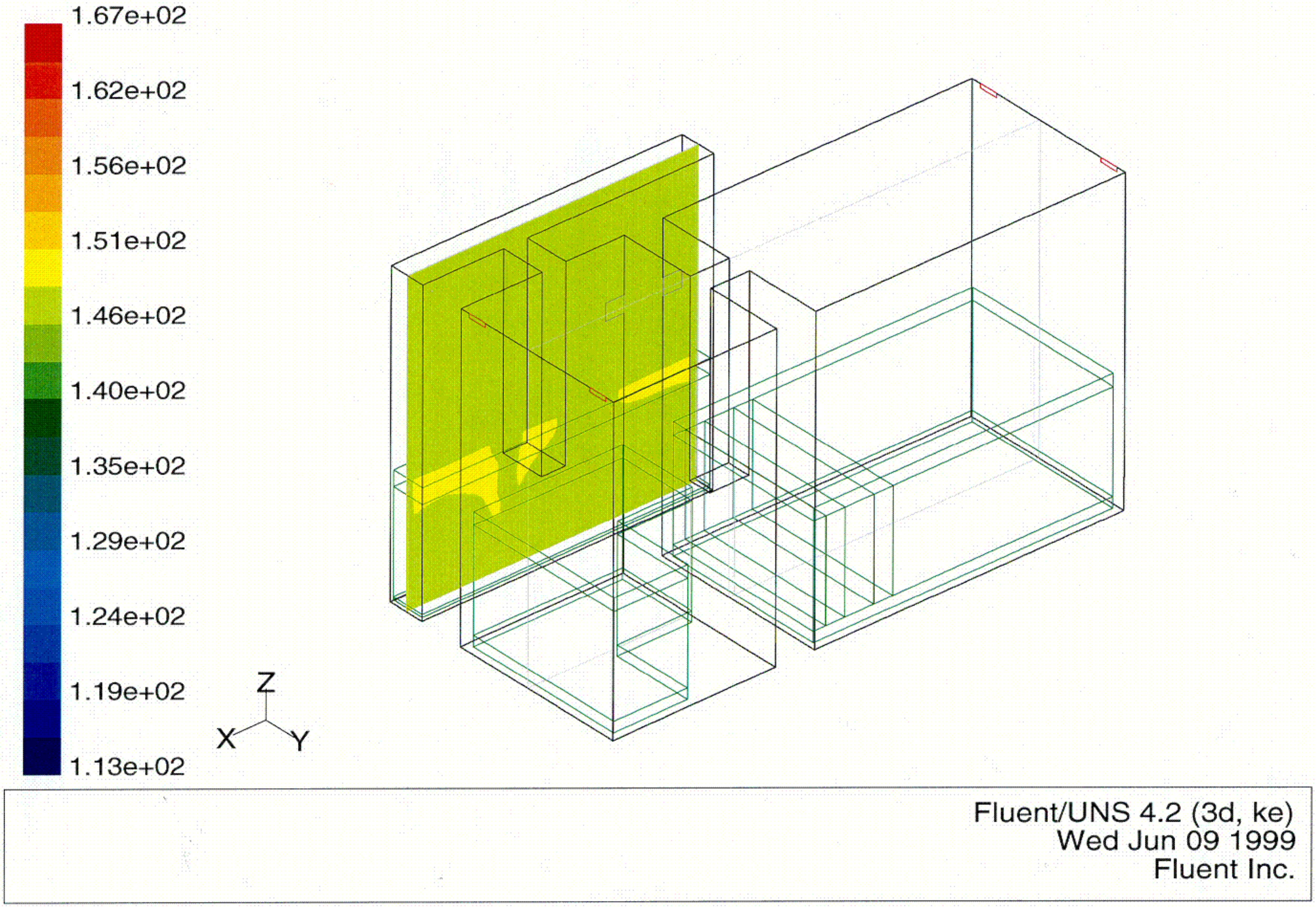

Figure 6: Temperature (degree F) contours in a plane through the middle of the Kewaunee transfer canal racks (Emergency Full Core Discharge Scenario)

C6

ATTACHMENT 3

Letter from M. L. Marchi (WPSC)

to

Document Control Desk (NRC)

Dated

August 7, 2000

Structural Issues

Requests For Additional Information

Structural Issues

Question 1

You indicated in Chapter 6 of the Reference cited below that the structural analyses of the new Transfer Canal spent fuel racks for the required loading conditions were performed in compliance with the US NRC Standard Review Plan (SRP) and the former US NRC Office of Technology (OT) Position Paper related to spent fuel storage. With respect to your structural analyses using the DYNARACK computer code:

(a) Explain how the target (design basis) response spectra (referred to in Section 6.4 of the Reference) were obtained.

(b) Provide the analyses results that show that the design criteria related to kinematic stability (i.e., safety factors against rack overturning) described in Chapter 2 of the Reference have been satisfied.

Response to Question 1(a)

The target (design basis) response spectra are the original licensing basis in-structure response spectra (ISRS) developed for the Kewaunee site and described in the KNPP Updated Safety Analysis Report (USAR).

The ISRS were developed from a free field ground motion spectrum developed specifically for the Kewaunee site by the consulting firm of Dames and Moore in 1968. The design basis earthquake (safe shutdown earthquake) was based upon a maximum horizontal ground acceleration of 0.12g. The ground motion spectrum was used as the seismic input at grade level to develop the IRSR for various floor elevations in the reactor, auxiliary, turbine, and screenhouse buildings.

Use of the plant licensing basis floor response spectra is allowed by the SRP and OT Position Paper, provided that the appropriate damping factors are used. For the rack analyses, a damping factor of 1.0 percent was used, which is in accordance with the requirements specified in the KNPP USAR.

The KNPP ISRS were previously reviewed and approved by the USNRC for resolution of the USI A-46 program at KNPP. The ISRS were classified as conservative design response spectra (as opposed to median-centered design response spectra). Reference [1] provides documentation of the USNRC review.

Response to Question 1(b)

Kinematic stability of the new storage racks is assured by observation of the layout given in Figure 1.2.1. As stated in Section 6.6.1 of Holtec report HI-992208, "The installed dimensions between the rack periphery and the adjacent Transfer Canal wall or rack precludes rack overturning."

The maximum rack displacement under any condition is 0.317 inches, as stated in Section 6.8.1. Racks do not impact the adjacent walls or racks under any conditions. In order for the rack to tip over, the rack centroid must be displaced beyond the rack periphery. This would translate into top of rack displacements exceeding the width of the rack (about 42 inches), since the centroid is approximately located at one-half of the rack height. Conservatively neglecting the presence of the wall, the factor of safety for overturning is given by:

42" / 0.317" = 132.

Question 2

You indicated in the Reference that the design conditions described in SRP 3.8.4 and American Concrete Institute (ACI) Code 349-85 were used as guidance in the calculations of the spent fuel pool (SFP) capacity. With respect to the SFP capacity calculations using the ANSYS computer code presented in Chapter 8 of the Reference:

(a) Describe how you obtained the design loads (including those due to seismic excitation) acting on the new wall to be built in the Transfer Canal, and provide a summary of the results of the structural analysis (including the safety factors) of the new wall.

(b) Explain how the interface between the liner and concrete slab is modeled, and also how the liner anchors are modeled; explain how such modeling accurately represents the real structural behavior.

Response to Question 2(a)

The partition wall is a dividing wall whose purpose is to divide the transfer canal (TC) into two independent sections. The proposed amendment permits the installation of spent fuel racks into one portion of the divided transfer canal.

The design of the new partition wall was conducted under the design loads and load combinations mandated by the NUREG-0800 SRP 3.8.4 specifications that are bounding for those load combinations mandated by the governing ACI Code.

The analysis considered the following design load categories: Dead Loads, OBE and SSE Seismic Loads, Normal Operating and Accident Temperature Loads. Besides the loads that act directly on the new wall, the loads applied to the new wall from the existing structure (across the interface) were also considered in a conservative manner. Conservatively assuming that the portion of the divided TC that remains unracked is empty of water, the following two limiting load combinations were deemed bounding and were analyzed:

1. Normal Operating (1.4 D + 1.9 E)
2. Accidental (D + Ta + E')

Dead Loads (D) include: wall self weight, hydrostatic pressure from the new TC pool, weight of the racks fully loaded with spent fuel.

The stress resultants under OBE and SSE Loads (E and E') are obtained from a quasi-static analysis, considering: inertia and sloshing pressures from pool water, hydrodynamic pressure from the motion of the new racks, inertia loads from the new wall and existing structure, and loads on the pool slab from the racks seismic response. The analysis assumes that peak input values from the N-S, E-W, and vertical components of the specified earthquake excitation act simultaneously.

The thermal stress analysis under Accidental Temperature Load (Ta) considers the temperature increase in the new TC pool from 50°F during the construction to water boiling temperature of 212°F and in the existing pools from normal operating conditions temperature of 150°F to the boiling temperature of 212°F.

Structural tees, fabricated from 1-inch thick steel plates, will be affixed to the existing walls using anchor bolts. These "shear keys" provide the structural connection between the new wall and the existing TC walls. The structural analyses reflect the fact that the shear keys transfer no tensile stresses.

The structural analysis of the new wall together with its connection with the existing structure yielded the following conservative estimates of safety factors:

Normal Operating Conditions - Load Combination 1.4 D + 1.9 E			
Member	Critical Force/Moment	Capacity	Safety Factor
Vertical Cross Section Bending	12.0×10^{3} lb. in.	94.7×10^{3} lb. in.	7.89
Horizontal Cross Section Bending	75.7×10^{3} lb. in.	120×10^{3} lb. in.	1.58
Contact Cross Section Bearing	677 lb.	14.2×10^{3} lb.	14.03
Contact Cross Section Shear	677 lb.	6.9×10^{3} lb.	10.17
Contact Cross Section Bending	1.69×10^{3} lb. in.	51.2×10^{3} lb. in.	30.26
Shear Key Shear	677 lb.	18.9×10^{3} lb.	27.99
Shear Key Bending	1.69×10^{3} lb. in.	7.7×10^{3} lb. in.	4.58
Anchor Bolt Shear *	677 lb.	2.84×10^{3} lb.	4.19
Extreme Environmental Conditions – Load Combination D + Ta + E'			
Member	Critical Force/Moment	Capacity	Safety Factor
Vertical Cross Section Bending	87.5×10^{3} lb. in.	94.7×10^{3} lb. in.	1.08
Horizontal Cross Section Bending	109.0×10^{3} lb. In.	120×10^{3} lb. in.	1.10
Contact Cross Section Bearing	1.84×10^{3} lb.	14.2×10^{3} lb.	5.17
Contact Cross Section Shear	1.84×10^{3} lb.	6.9×10^{3} lb.	3.75
Contact Cross Section Bending	4.59×10^{3} lb. in.	51.2×10^{3} lb. in.	11.14
Shear Key Shear	1.84×10^{3} lb.	18.9×10^{3} lb.	10.31
Shear Key Bending	4.59×10^{3} lb. in.	7.7×10^{3} lb. in.	1.69
Anchor Bolt Shear *	1.84×10^{3} lb.	2.84×10^{3} lb.	1.54

* The shear capacity of the anchor sleeves is conservatively neglected.

Response to Question 2(b)

The pool liner is not included in the overall 3-D ANSYS structural model of the spent fuel pool. Any contribution to the pool structural support by the thin liner is conservatively neglected. The stress analysis of the liner is considered in a separate analysis focused on the in-plane stress distribution. The liner in the Kewaunee transfer canal is assembled from austenitic steel plates which are seam welded along their contiguous edges resulting in a sealed container geometry to hold the pool water. The seam weld lines are also locations of anchorage. The integrity analysis of the pool liner consisted of the following evaluations:

(i) Ensure that the in-plane stresses in the liner from the fuel rack pedestals during the seismic event would not cause rupture in the liner from a single load application.

(ii) Ensure that repetitive loading during a seismic event will not lead to fatigue failure in the liner (1 SSE and 20 OBE's occurring in sequence is the design basis)

To evaluate the stress field in the liner, it is modeled as a 2-D plate, which is fixed along its edges to simulate the weld seams. The liner anchors are assumed to be rigid, and therefore, are not explicitly modeled. It is conservative to assume rigid anchorage, as any displacements in the connections would reduce the calculated liner stresses. A bounding geometry was utilized wherein the anchor lines are conservatively assumed to be nearest to the pedestal location. The finite element solution evaluated the stress distribution at the line of support representing the weld seam.

Question 3a

Section 7.5.2 "Deep Drop Events" in Attachment 5 of the Reference states that the deep drop through an exterior cell does produce some deformation of the baseplate and localized severing of the baseplate/cell welds. You further state that the fuel assembly support surface is lowered by a maximum of 1.068 inches, which is less than the distance of 6 inches from the baseplate to the liner. Provide the design limit of the allowable deformation of the baseplate, and discuss the impact of the localized severing of the baseplate/cell wall welds on the integrity of the racks and the fuel assemblies.

Response to Question 3a

The allowable deformation of the baseplate due to the deep drop event is limited to the nominal distance between the baseplate and the liner plate, which is 6". This limit is to prevent the baseplate from contacting the liner plate. The baseplate also must not experience global failure or puncture as a result of the deep drop event. Figure 7.5.2 has shown that the above acceptance criteria are met for the postulated deep drop event. The maximum vertical deformation of the baseplate is calculated to be only 1.068", which is much smaller than the 6" limit.

The localized severing of the baseplate/cell wall welds will not lead to adverse hydraulic and criticality consequences and will not degrade the overall structural integrity of the rack module. The finite-element analysis demonstrates that the effect of the base plate deformation is confined to the impacted cell and the cells that are directly adjacent. Proper support of the stored fuel assemblies will remain following the localized deformation/weld severing due to the rigidity provided by the interconnection of module cells in the Holtec rack design.

Question 3b

In the same section on Deep Drop Events cited above, you state that the deep drop event wherein the impact region is located above the support pedestal produces a maximum stress less than the failure limit in the liner. You further state that the maximum compressive stress in the concrete slab is smaller than the failure limit of the confined concrete. Provide the maximum stresses in the liner and in the concrete slab, and the failure limits of these stresses, citing the references which give these failure limits.

Response to Question 3b

For the deep drop event wherein the impact region is located above the rack pedestal, the maximum Von Mises stress in the liner is shown to be 25.32 ksi in Fig. 7.5.3. The failure stress of the stainless steel liner is 66.2 ksi, which can be found in Reference [2]. The maximum compressive stress in the concrete is 15.75 ksi as shown in Fig. 7.5.4. With regards to Kewaunee spent fuel pool structure, the upper stratum of this concrete slab supporting the stainless steel liner, which is laterally confined and simultaneously compressed from the interior of the pool water pressure, exhibits a tri-axial compressive stress behavior, which reduces the tendency of internal cracking. Reference [3] provides a plot of stress-strain curves for concrete subjected to tri-axial compression. Based on the stress-strain curves in Reference [3], the failure limit of the concrete is 20.22 ksi. This stress-strain plot has been used and accepted by the USNRC as input in the drop accident analyses for several different nuclear plants including Union Electric's Callaway Plant and Wolf Creek Plant and Commonwealth Edison's Byron and Braidwood plants.

References:

[1] Kewaunee Nuclear Power Plant – Safety Evaluation Report for USI A-46 Program Implementation (TAC No. M69455), dated April 14, 1998.

[2] ASME Boiler and Pressure Vessel Code, Section II - Part D, Properties.

[3] Park, R. and Paulay, T., "Reinforced Concrete Structures," John Wiley and Sons, 1975.