WOLF CREEK
NUCLEAR OPERATING CORPORATION

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ET 98-0070

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

Reference:
1) Letter ET 98-0009, dated March 20, 1998, from R. A. Muench, WCNOC to USNRC
2) Letter ET 98-0041, dated May 28, 1998, from R. A. Muench to USNRC
3) Letter ET 98-0048, dated June 30, 1998, from R. A. Muench to USNRC
4) Letter dated July 23, 1998 to O. L. Maynard from USNRC

Subject: Docket No. 50-482: Response to Request for Additional Information (RAI) Related to Proposed Revision to Increase the Spent Fuel Pool Storage Capacity

Gentlemen:

Attached is Wolf Creek Nuclear Operating Corporation's (WCNOC) response to Reference 4, NRC request for additional information (RAI). Reference 1 provided the original submittal of an amendment request to revise the Wolf Creek Generating Station (WCGS) technical specifications to support modification to increase the Spent Fuel Pool capacity at WCGS. Reference 2 transmitted revisions to original submittal Chapters 4 and 5 of the Licensing Report to reflect a re-assessment of the proprietary classification of proprietary versus non-proprietary material. Reference 3 provided WCNOC's response to a Request for Additional Information dated June 4, 1998. The Attachment to this letter provides WCNOC's response to Reference 4.

If you have any questions concerning this response, please contact me at (316) 364-8831, extension 4034, or Mr. Michael J. Angus, at extension 4077.

Very truly yours,

Richard A. Muench

A001

RAM/rlr

Attachment

cc: V. T. Cooper (KDHE), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
B. A. Smaildridge (NRC), w/a
K. M. Thomas (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Richard A. Muench, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By _____
Richard A. Muench
Vice President
Engineering

SUBSCRIBED and sworn to before me this 28th day of August, 1998.

_____
Notary Public

NOTARY PUBLIC
STATE OF KANSAS
CINDY NOVINGER
My Appt. Exp. 7/8/2002

Expiration Date July 8, 2002

Response to Request for Additional Information
Spent Fuel Pool Rerack License Amendment Request

Question 1:

The May 28, 1998, submittal states that administrative controls will be implemented to ensure that dose rates external to the building will meet the limits for the general public. It also states that additional controls will be implemented to ensure that interior building dose rates are maintained within the ranges of the specified zone designations. Describe what additional controls you will take to ensure that the current radiation zone designations are not exceeded. In addition to administrative controls, discuss the use of any bridge movement interlocks that may be used to control the placement of fuel assemblies in the fuel racks.

Response:

Administrative controls will be implemented to define acceptable storage locations for spent fuel assemblies to ensure that specified radiation zone designations are not exceeded. The administrative controls will be based on calculations and will be documented in the form of plant procedures associated with special nuclear material control. The administrative controls will establish minimum cooling times for spent fuel assemblies prior to storage in designated locations. In addition, routine radiological dose surveys in accordance with Health Physics procedures will record dose rates to ensure zone designations are not exceeded. Administrative controls will be adequate to prevent placement of fuel assemblies into unauthorized storage locations, therefore bridge movement interlocks will not be required.

Calculations have been performed to provide one set of administrative controls which may be implemented to maintain zone designations at the same level as the current zone designations (with the exception of the cask loading pit and cask washdown pit areas). The zone designation for the cask washdown pit area (USAR Figure 12.3-2) will change from "B" (<2.5 mrem/hr) to "E" (>100 mrem/hr) based on the calculated dose rates associated with the most limiting case analyzed. Access controls will be implemented for the new Zone "E" area in accordance with Health Physics procedures. The zone designation above the pool surface in the cask loading pit will be changed from zone "B" to zone "C," which matches the current zone designation above the Spent Fuel Pool given on USAR Figure 12.3-2.

Question 2:

Your submittal states that you will install three spent fuel pool (SFP) racks (with a combined capacity of 279 fuel assemblies) in the cask loading pit in a later campaign. Discuss how the presence of spent fuel in these racks will affect the dose rates in accessible areas adjacent to the cask loading pit both during storage and movement of the spent fuel assemblies into and out of the pit.

Response:

The increase in dose rates due to fuel stored in the cask loading pit was considered by calculating dose rates at discrete locations around the pit. Calculations were performed to establish storage limitations

which would confirm dose rates within the current zone designation ranges, except as established by the response to Question 1. The supplemental calculations were based on conservative fuel parameters, since most of the fuel in the cask loading pit is considered to be cooled only 100 hours subsequent to reactor discharge. This is conservative because the cask loading pit will not contain any freshly discharged fuel. These storage limitations will be implemented through administrative controls based on establishing minimum cooling times for designated storage locations prior to storage. These controls provide a greater distance from the accessible locations to fuel cells which have no cooling time restrictions, and/or a barrier shield using fuel which has had a lengthy cooling time.

The calculated dose rates based on conservative fuel parameters and the worst case configurations allowed by the storage location limitations are as follows:

a) At the outside of the Fuel Building wall surface at a point directly east of the spent fuel location in the cask loading pit, the dose rate is calculated to be 0.08 mR/hr.

b) At a point along the wall in the truck bay located directly south of the cask loading pit, the dose rate is calculated to be 2.14 mR/hr.

The dose rates listed above fall below the designated limits. At the fuel operating deck (elevation 2047'-6") the approximate 25 feet of water covering the active fuel region, coupled with the adjacent concrete walls, provide more than adequate shielding to maintain dose rates above the water surface below the zone C limit of 10 mR/hr. The dose on the fuel operating deck due to fuel stored in the cask loading pit will be less than or equal to the dose due to fuel stored in the Spent Fuel Pool, which has similar distance and shielding to the areas accessible to personnel.

The dose due to fuel in transit remains unchanged from the previous conditions. Fuel movement within the cask loading pit has taken place previously since the fuel reconstitution station is located within the pit as discussed in USAR Section 9.1.4.1.1. The elevation at which fuel is raised by the fuel handling machine remains unchanged as does the nominal water depth. Therefore, the shielding provided by the Spent Fuel Pool or cask loading pit water is unchanged and the dose rate to the operator and areas surrounding the Spent Fuel Pool or cask loading pit will not change.

Question 3:

Describe any sources of high radiation that may be in the Wolf Creek Plant SFP during diving operations to remove the old SFP racks and install the new racks. Discuss what precautions (such as use of TV monitoring, tethers, etc.) will be used to ensure that the divers will maintain a safe distance from any high radiation sources in the SFP. Describe how you plan to monitor the doses received by the divers during the reracking operation (e.g., use of dosimetry, alarming dosimeters, remote readout radiation detectors).

Response:

Some sediment is expected to have settled at the bottom of the Spent Fuel Pool and cask loading pit. Considering Wolf Creek has had fuel leakage in the past, it cannot be ruled out that highly radioactive particles, such as fragments of fuel pellets or crud from fuel clad surfaces, may be located on surfaces within the Spent Fuel Pool or cask loading pit. Other sources of high radiation in the Spent Fuel Pool include the spent fuel assemblies and associated inserts, a fuel rod storage basket for failed fuel rods, a rod segment storage basket, and a trash basket. In addition, sources of high radiation in the cask loading pit include thimble plugs and miscellaneous components removed from fuel assemblies. Potential sources of high radiation due to contamination include the spent fuel storage racks, a trash rack containing miscellaneous loose parts, the new fuel elevator, fuel handling tools, pool walls, and submerged piping or pipe support surfaces. An underwater vacuum cleaner system will be used in conjunction with radiation surveys to ensure appropriate areas are vacuumed as part of dive preparations.

Wolf Creek Health Physics is aware of the potential for radiation exposure challenges during diving operations in the Spent Fuel Pool or the cask loading pit. Diving operation procedures describe the Health Physics coverage associated with each dive. The procedure requires pre-dive surveys with at least two independent survey instruments, with specified agreement criteria between the survey results. The procedure requires the use of remote reading electronic dosimeters on the diver, and the readout from these dosimeters be continually monitored during the dive. The procedure also requires that the diver have available an operable survey instrument so that he can survey unknown items. The readout is maintained at the dive control station and monitored by a Health Physics Technician. Constant communications is required between the dive control station and the diver.

The physical methods of controlling the divers spatial location to keep the diver away from high radiation sources will depend upon the considerations of each dive, and will be determined by the ALARA review in consultation with the dive controller. The primary method of control is to adequately brief the diver on the particular job and provide constant visual oversight of the diver from the dive control station. The use of visual barriers such as air bubbles, ropes, or orange safety netting or the use of physical barriers such as tethering the diver umbilical and other control measures will be implemented if the situation is determined to warrant it by the ALARA group or the dive controller.

The diver will be in continuous voice communication with the dive controller/Health Physics personnel. In the event that an unexpectedly high radiation field/hot particle is encountered by the diver, he will be verbally instructed to take the appropriate corrective action. Positive control of the diver will be maintained, at all times, through use of an individual on the surface maintaining a diver safety line.

Radiological surveys are required for the dive area including entrance and exit travel routes to and from the Spent Fuel Pool. These surveys will consist of surveying the general work area and specific components with an underwater probe capable of detecting expected radiation levels. Radiological surveys are required after any movement of irradiated hardware and prior to diving after such movement.

During the planning stage, the sequence of work will be evaluated to ensure safe dive areas are established for any diving activity. Sketches of rack layouts and positions of irradiated hardware will be reviewed by site personnel, prior to each dive, to ensure sufficient space has been provided between the diver and irradiated hardware. A pre-job brief will be conducted with the diver to discuss radiological conditions and work scope prior to each dive.

Every dive situation will be evaluated to ensure the safety of the diver.

Question 4:

Provide the calculated 30 day doses (thyroid and whole body) to the control room operator as a result of a fuel handling accident (occurring both in the fuel handling building and in containment). Describe the calculational method used to arrive at these doses and include all assumptions. Describe any differences that may exist between control room isolation following a fuel handling accident and control room isolation following a LOCA and describe how these differences would affect operator 30 day accident doses.

Response:

The potential 30 day radiological consequences to a control room operator resulting from a postulated fuel handling accident in the containment, with containment personnel airlock doors remaining open during core alterations or fuel movement, have been calculated to be 1) 9.7 rem Thyroid and 2) 2.88E-2 rem whole body.

The potential 30 day radiological consequences to a control room operator resulting from a postulated fuel handling accident in the fuel building i not explicitly analyzed as the fuel handling accident analysis in containment is bounding due to 1) the fission product inventory being larger, due to the number of assumed damaged rods being 20% greater and 2) the release path excluding filtration system removal of radionuclides prior to their release to the atmosphere and control room.

The calculational methods used to assess the potential radiological consequences, including the buildup of activity in the control room and the integrated doses to control room personnel, are described in WCGS USAR Sections 15.7.4, 15A and the associated explicitly referenced tables and sections within the USAR.

The assumptions postulated are consistent with the assumptions of Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," which are summarized below:

a) The accident is postulated to occur 100 hours after shutdown, which is the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during this time period is taken into consideration.
b) The minimum water depth between the top of the damaged fuel rods and the refueling pool surface is 23 feet.

c) The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All fuel rods contained in the dropped assembly are assumed to be damaged in both the fuel handling accident in the containment and in the fuel building analyses. In addition, the dropped assembly is assumed to damage an additional 20 percent of the rods of another assembly in the fuel handling accident in containment analysis.
d) The values assumed for individual fission product inventories in the damaged assembly were calculated based upon a radial peaking factor of 1.65.
e) Damaged rods are assumed to release their gap activities. The gap activity released to the refueling pool from the damaged fuel rods consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine contained in the fuel rods at the time of the accident.
f) The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).
g) The effective pool decontamination factor for iodine is taken as 100 (i.e., 99 percent of the total iodine released form the damaged rods is retained by the pool water).
h) For the fuel handling accident in containment analysis, the gaseous effluent escaping from the refueling pool in containment is assumed to be released to the environment through the open personnel airlock and the adjacent auxiliary building without mixing in the surrounding atmosphere.
i) For the fuel handling accident in containment analysis, the auxiliary building atmosphere is normally exhausted through filter absorbers designed to remove iodine. However, no credit is taken for iodine removal by the atmosphere filtration system filters.
j) In both the fuel handling accident in the containment and the fuel building analyses, the radioactive material that escapes from the respective buildings is assumed to be released from the building over a two-hour time period.

Actuation of the control room emergency ventilation system by a Control Room Ventilation Isolation Signal (CRVIS), places the system in the emergency mode of operation. The LOCA and fuel handling accident analyses use similar assumptions regarding the timing of control room isolation. None of the analyses rely on actuation of the control room HVAC radiation monitors.

Actuation of the control room emergency ventilation system is initiated by a CRVIS which receives input from the radiation detectors, the fuel building ventilation isolation signal, the containment isolation phase A, the high containment atmosphere radiation, the containment purge isolation signal, or manually. The signal which initiates a CRVIS depends upon the accident. Following a LOCA, a CRVIS is initiated by the containment isolation phase A signal which is initiated by a safety injection signal. For the fuel handling accident in containment, a CRVIS is initiated by a containment atmosphere radiation or containment purge isolation signal. For a fuel handling accident in the fuel building, a CRVIS is initiated by a fuel building ventilation isolation signal.

The differences in the radiological consequences to control room personnel depend primarily upon the nature of the accident, the LOCA being the most limiting. The potential radiological consequences from containment and Engineered Safety Features component leakage to a control room operator as a result of a LOCA, have been calculated to be 1) 22.83 rem Thyroid and 2) 0.24 rem Whole Body.

Question 5:

Verify that all of the assumptions used in the Final Safety Analysis Report (FSAR) fuel handling accident analysis are still applicable. Also verify that the resulting postulated thyroid and whole body doses at the Exclusion Area Boundary and Low Population Zone as a result of a fuel handling accident are still valid.

Response:

The assumptions used in the USAR fuel handling accident analysis remain applicable with the major exceptions being, a) the current Spent Fuel Pool enrichment limit is 4.45% and b) the Spent Fuel Pool rerack enrichment limit is increased to 5.0%. The change in the core inventory of fission products is not sensitive to fuel enrichment and any attendant extended burnup, due to increased enrichment, is not expected to change calculated offsite dose results significantly (WCAP-10125-P-A).

The assumptions postulated in the calculation of the radiological consequences of a fuel handling accident are consistent with the assumptions of Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," and those delineated in response to Question 4 above.

The postulated radiological consequences, both thyroid and whole body at the Exclusion Area Boundary and Low Population Zone, of a fuel handling accident (USAR Table 15.7-8) are determined to be valid. This determination is based upon an examination of the analysis in support of the rerate to 3579 MWth (Technical Specification Amendment No. 69) and the analysis of both doors of the containment personnel airlock remaining open during core alterations or fuel movement (Technical Specification Amendment No. 95), respectively.

The calculated doses for the exclusion area, low population zone, and control room may change slightly from those presented in the current licensing basis as a result of implementing the Spent Fuel Pool rerack program, but, in all cases will remain below applicable regulatory limits. A new source term representing the enrichment limit of 5.0% is being developed. A calculation of the doses for the exclusion area, low population zone, and control room to demonstrate compliance to the applicable regulatory limits will be completed prior to installation of the racks.

Question 6:

Discuss the shipment and disposal for the old spent fuel rack modules.

Response:

The racks will be rinsed with demineralized water while being removed from the pool to remove contaminants to an acceptable level. Administrative controls will prevent the demineralized water from diluting the pool below acceptable boron concentration limits. After removal from the pool the racks will be bagged, sealed, down-ended, and

placed into a special DOT approved shipping container. The rack will be braced inside the container, prior to sealing the container, to prevent shifting during transit. The container and enclosed rack will be shipped on a flatbed truck to Manufacturing Science Corporation (MSC) in Oak Ridge, Tennessee for disposal. Health Physics personnel will monitor the packaging prior to shipment to assess dose rates and to ensure the packaging will prevent dispersal of contaminants. The shipments will be made in accordance with Health Physics and Radwaste department procedures governing shipment of radioactive material / waste and will meet the applicable requirements of 49 CFR and 10 CFR 20.

Question 7:

Discuss how the storage of the additional spent fuel assemblies in the Wolf Creek Station SFP will affect the releases of radioactive gases (specifically Kr-85, I-131 and tritium) from the SFF.

Response:

Release of radioactive gases by WCGS will remain a small fraction of the limits of 10 CFR 20.1301 and the design objectives of Appendix I to 10 CFR 50 following the implementation of the proposed modification to increase the capacity of the WCGS Spent Fuel Pool. This conclusion is based on the following supporting statements:

a) The half lives of short lived isotopes such as I-131 are short in comparison to fuel cycle length, therefore, short-lived nuclides are present only in freshly offloaded fuel. The quantity of freshly offloaded fuel placed into the Spent Fuel Pool each refueling outage is not affected by the number of spent fuel assemblies being stored in the Spent Fuel Pool. Therefore, the inventory of I-131 in the Spent Fuel Pool will not be affected by the increased Spent Fuel Pool capacity.

b) Inventories of long-lived fission products (e.g. Kr-85, I-129, and ternary tritium) in spent fuel assemblies will decrease slowly within individual fuel assemblies over years in storage. Therefore, an increase in the number of spent fuel assemblies stored in the Spent Fuel Pool would increase the total Spent Fuel Pool inventory of these radionuclides. However, these radionuclides are not released in significant amounts from the stored fuel to the Spent Fuel Pool water, even for failed fuel since fuel pellet temperature of stored fuel is not high enough to create sufficient gas pressure in the gap to overcome the static pressure of the Spent Fuel Pool water.

c) The radioactivity in the Spent Fuel Pool water would be maximized when the Spent Fuel Pool and RCS are connected during refueling outages since the radioactivity originates in the RCS and is introduced into the Spent Fuel Pool.

d) The increased number of spent fuel assemblies in storage will raise the heat load on the Spent Fuel Pool and could result in an increase in the evaporation rate. Other than a small amount of tritiated water released by evaporation, Spent Fuel Pool radionuclides are non-volatile and, consequently, are not released from the pool water. The increased evaporation rate of tritiated water would result in an increase in gaseous tritium released in the Wolf Creek effluents. However, the discharge of gaseous

radioactive effluents will continue to be a small fraction of the limits of 10 CFR 20.1301 and the design objectives of Appendix I to 10 CFR 50.

Question 8:

Discuss how the storage of the additional spent fuel assemblies will affect the releases of radioactive liquids from the plant.

Response:

The number of spent fuel assemblies in storage does not affect the release of radioactive liquids from the plant. The contribution of radioactive materials in the Spent Fuel Pool water from the stored assemblies is insignificant relative to other sources of activity, such as the reactor coolant system. The volume of Spent Fuel pool water processed for discharge is independent of the number of fuel assemblies stored in the Spent Fuel Pool.

Question 9:

Discuss your plans to use a vacuum to remove any crud or other debris from the floor of the SFP before and during the SFP reracking project. Also describe any radiation surveys that will be performed (from the pool rim or by divers in the pool to map dose rates in the SFP.

Response:

During reracking, a Tri-Nuc underwater filtration unit with associated filters will be used to vacuum the Spent Fuel Pool. A backup system will also be available. Vacuuming will consist of vacuuming crud after a series of "old" racks have been removed. Vacuuming will also occur in areas in which the diver may have to enter, provided the dose rates in the area warrant vacuuming.

Performance of radiation surveys is discussed in the response to question 3.

Question 10:

Is full core off-load the general practice for planned refueling outages?

Response:

Yes. A full core off-load is the general practice for planned refueling outages.

Question 11:

If full core off-load is the general practice, then the corresponding heat load is the one considered for normal operation. Table 5.8.1 states that for the postulated full core discharge, the maximum allowable heat load is 63.41 MBtu/hr, which corresponds to a maximum bulk SFP temperature of 170°F. This limit exceeds the American Concrete Institute (ACI) Standard 379, which states in part, "for normal

operation or any other long term period, the [SFP] temperatures shall not exceed 150 degrees." If the planned refueling is generally a full core off-load, provide an evaluation on how long the SFP temperature would be above 150°F, and the justification for why the 170°F limit is acceptable to meet the intent of ACI-379.

Response:

It is assumed that the "(ACI) Standard 379" referenced in the question represents a typographical error and is intended to refer to ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349)." Appendix A, Section A.4.1 states, "The following temperature limitations are for normal operation or any other long term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are allowed to have increased temperatures not to exceed 200°F." This short section of the Standard does not include any detailed definitions for "normal operation" or "long term period." However, it is implied that normal operations are considered to occur over long term periods. The ACI implied intention of normal operations appears to differ significantly from the conditions required to produce the 170°F bulk pool temperature. As stated in response to Question 10, full core offload is the normal practice during refueling outages. However, the 63.41 MBtu/hr decay heat load limit is not considered to be a long term normal operation heat load. The heat load of 63.41 MBtu/hr corresponds to a steady state bulk pool temperature of 170°F when the heat balance is calculated assuming one train of cooling in operation, and the design basis maximum coolant water heat exchanger inlet temperature of 105°F. Using this conservatively calculated decay heat load for a full core offload, it has been determined that the maximum time period that the pool temperature would be above 150°F is less than 9 days per fuel cycle. Therefore, the 170°F bulk pool temperature results in concrete temperatures exceeding 150°F for an acceptable period of time (i.e., 9 days) and in only a portion of any concrete cross-section (i.e., local areas), as allowed by the ACI Code.

The 170°F bulk pool temperature is considered acceptable for the following reasons:

a) As noted by the question, the 170°F temperature represents a peak temporary condition. During the time of this bulk pool temperature peak, the concrete temperature will not be homogeneously elevated to 170°F because of its thermal inertia. The concrete elevated temperature would be based on a thermal gradient occurring between the bulk pool temperature and the temperature at the outer surface of the concrete pool enclosure. This gradient passes through the water/liner interface film resistance, the liner plate, any small air pockets between the liner plate and the concrete surface, and finally the massive concrete cross-section itself. Therefore, only a portion of any particular concrete section is expected to exceed the temperature of 150°F, as allowed by the ACI Code.

b) The evaluation of the concrete pool structure considers the thermal gradient across sections, in accordance with the methods recommended by ACI-349, by considering the 170°F temperature in

the pool as a normal condition. In fact, the pool structure evaluation also considers the more extreme accident case of pool boiling with an even greater temperature condition.

c) It is common practice for the bulk temperature at many plants to exceed the recommended 150°F limit during short durations. In fact, the current licensing basis allows the bulk pool temperature to approach approximately 160°F, as stated in USAR Section 9.1.3.3.

Question 12:

Is there a procedure for performing the outage specific evaluation of heat load? What value is used for the maximum pool heat load?

Response:

As part of WCNOC's Reload Design Process as described in procedure AP 19E-001, "Reload Design and Safety Evaluation," the cycle specific Spent Fuel Pool heat load and bulk pool temperature for a full core offload case are calculated and documented in the Reload Safety Analysis Checklist (RSAC) to ensure that all Spent Fuel Pool design requirements under a full core off load condition continue to be met. Depending upon the cycle specific fuel loading strategy and the assumptions used, the cycle specific Spent Fuel Pool heat load and pool bulk fluid temperature will be calculated to confirm that the heat load is bounded by the maximum allowable heat load of 63.41 MBtu/hr, in order to ensure the calculated bulk pool temperature will remain below the established temperature limit.

The value used for the maximum pool heat load is based on the decay heat associated with a full core (193 assemblies) removed from the reactor following a reactor shutdown, plus the additional heat load from assemblies already stored in the Spent Fuel Pool.

Question 13:

Discuss sources and capacities of make-up water and the methods/systems (indicating seismic design category) used to provide make-up water, as well as the time needed to set up a path for make-up water to the SFP.

Response:

Technical Specification Surveillance Requirement 4.9.11 requires the water level in the Spent Fuel Pool be determined to be at least its minimum required depth (-27 inches on instrument EC LI-39A which is equivalent to 23 feet above the top of irradiated fuel assemblies) at least once per 7 days when irradiated fuel assemblies are in the Spent Fuel Pool. Surveillance Procedure STS CR-001, "Shift Logs for MODES 1, 2 and 3" requires logging the level every 8 hours with an administrative limit of -1.5 inches. If the Spent Fuel Pool water level is below the administrative limit, make-up to the Spent Fuel Pool is performed in accordance with procedure SYS EC-200, "Changing Level in the Spent Fuel Pool or Refueling Pool."

Procedure SYS EC-200 provides two sources for make-up to the Spent Fuel Pool depending on the Spent Fuel Pool boron concentration. The normal source of makeup water is from the Reactor Makeup Water tank via the Reactor Makeup Water pumps. The second source of makeup water (as

specified in this procedure) to the Spent Fuel Pool is from the Refueling Water Storage Tank via the Spent Fuel Pool cleanup pumps.

If a Spent Fuel Pool low level alarm (-10 inches) is received, procedure ALR 00-076D, "SFP LEV HILO" provides for actions to restore level. If Spent Fuel Pool level is stable, procedure SYS EC-200 is performed to restore level. If Spent Fuel Pool level is decreasing, off-normal procedure OFN KE-018, "Fuel Handling Accident" identifies five sources of makeup water for restoring level. This information is based on the current Spent Fuel Pool rack configuration.

The below table provides information on the various sources of make-up water to the Spent Fuel Pool.

WATER SOURCE	**CAPACITY** (gpm)	**PUMPS**	**SEISMIC** (pump/piping)	**Time to begin makeup** (min. approx.)
Reactor Makeup Water (RMW) Tank	20	RMW pump	No	30
Refueling Water Storage Tank (RWST)	300	SFP cleanup pumps	No	30
Recycle Holdup Tank (RHUT)	100	Recycle Evaporator feed pumps	No	90
Essential Service Water (ESW) System	25	ESW pump	Yes	30
Fire Main Water via fire hoses	200	Fire pump	No	30

Question 14:

The March 20, 1998, submittal stated that additional racks will be placed within the cask loading pit during a later campaign. When do you anticipate installing these cask loading pit racks? Would newly off-loaded fuel be placed in the cask loading pit? If not, what controls would be in place to ensure that newly off-loaded fuel would not be placed in the cask loading pit? How do you ensure that sufficient cooling is provided to the fuel located in the cask loading pit?

Response:

If permanent off-site storage options become available in a timely manner, racks may never be installed in the cask loading pit. Additionally, based on current projections, the reracked Spent Fuel Pool will provide sufficient capacity to accept a full core discharge at the end of the current operating license in March 2025. At that time there is projected to be 2353 fuel assemblies in storage, which is within the capacity of 2363 fuel assemblies in the reracked Spent Fuel Pool without the fuel storage racks in the cask loading pit. If required to support license renewal, Wolf Creek would anticipate the installation of racks in the cask loading pit prior to March 2025. An earlier installation date could be realized if discharge rates increase or additional storage flexibility is required.

Wolf Creek will use administrative controls to ensure that newly off-loaded fuel is not stored in the cask loading pit. The administrative controls will be documented in the form of plant procedures associated with special nuclear material control.

If spent fuel is stored in the cask loading pit it will be adequately cooled by the passive, buoyancy driven exchange of water with the Spent Fuel Pool. The sufficiency of cooling provided to the cask loading pit is demonstrated by the three-dimensional computational fluids model, which includes the cask loading pit and the interconnecting slot. The cask loading pit is assumed to contain the average background decay heat generation. The Computational Fluid Dynamics (CFD) results show that the flow rate through the interconnecting slot is sufficient to completely replace all the water in the cask loading pit in less than one hour. The temperature contours through the interconnecting slot show no substantial difference between the pool bulk temperature and the cask loading pit temperatures.

Question 15:

With regard to the FLUENT 3Dmodel, what are the model assumptions? Was a sensitivity study done with regard to the different physics and correlations options? Has the model been proven to be grid independent?

Response:

The following assumptions (12 total) for the CFD modeling have been paraphrased from the appropriate local temperature analysis calculation package.

1. The decay heat load is conservatively selected to bound earlier calculated decay heat load limits. This CFD evaluation is therefore performed for a higher heat load (65 MBtu/hr vs. 63.4 MBtu/hr) than is actually permitted.

2. The decay heat load contribution from the hottest 1/3 core in the Spent Fuel Pool is conservatively based on a bounding fuel rod heat flux value. This causes the decay heat from the hottest fuel assemblies to bound any actual condition.

3. The analysis uses the geometry of a Westinghouse 17x17 Standard fuel assembly. Of the fuel assembly types listed in the Joint Specification for Reracking, this fuel assembly has the lowest cell free flow area. This conservatively maximizes the hydraulic resistance and resulting local temperatures.

4. All storage locations are assumed to be 50% blocked at the top of the racks, thereby increasing the hydraulic resistance of the rack cells. This will conservatively bound any realistic cell blockage scenario.

5. All pedestal cells, which have no baseplate holes and must, therefore, rely on the cell wall side holes, are assumed to be located together in the Spent Fuel Pool. This highly conservative assumption creates a large flow restriction that affects a relatively large region of the Spent Fuel Pool, substantially worsening the local flow and temperature fields in that region.

6. The calculated hydraulic resistance parameters are conservatively worsened to bound any small deviations in fuel assembly geometry and rack construction tolerances. The two calculated parameters, permeability and inertial resistance factor, are worsened by 15% and 50%, respectively.

7. The assemblies from the hottest 1/3 core are assumed to be located in the region occupied entirely by pedestal cells. This conservatively locates the majority of the decay heat load in the region with the highest hydraulic resistance. This is an extremely conservative assumption as the inertial resistance in this region is nearly twenty-times greater than that for non-pedestal cells.

8. The minimum bottom plenum height in the Spent Fuel Pool is used for both the Spent Fuel Pool and the cask loading pit, even though the cask loading pit bottom plenum is actually much larger. This conservatively maximizes the hydraulic resistance and resulting local temperatures.

9. Along each wall of the Spent Fuel Pool and the cask loading pit, the minimum gap is used for all locations along that wall. Localized larger rack-to-wall gaps (i.e. near transfer canal gates) are neglected. This conservatively minimizes the downcomer area, thereby maximizing the hydraulic resistance and resulting local temperatures.

10. The downcomer spaces between racks are neglected. This conservatively minimizes the downcomer area, thereby maximizing the hydraulic resistance and resulting local temperatures.

11. Bounding minimum pool dimensions in the east-west and north-south directions are used instead of the nominal pool dimensions. This conservatively minimizes the downcomer area, thereby maximizing the hydraulic resistance and resulting local temperatures.

12. The total peaking factor is applied to the heat flux in the fuel cladding temperature calculation. This conservatively increases the decay heat generation rates, thereby maximizing the resulting fuel clad temperatures.

With all CFD codes, the ability to select appropriate mathematical models to correctly evaluate physical phenomena is based largely on knowledge gained through firsthand experience in modeling the phenomena. The FLUENT code is no exception. Once an acceptable combination of mathematical models, boundary conditions and discretizing schemes are discovered, however, similar situations can be evaluated using the same combinations.

Holtec's thermal analysts have developed a substantial amount of expertise with the use of the FLUENT code to evaluate Spent Fuel Pool thermal-hydraulic phenomena, particularly buoyancy-driven flows in porous media. The mathematical modeling options selected by the analysts reflect the experience gained through performing many analyses.

The computational burden associated with modeling of this class of problems is substantially increased by the range of length scales involved. While a typical pool has dimensions on the order of 40 feet, rack-to-wall gaps can range from a few inches to less than an inch.

Since buoyancy forces dominate the flow field in a spent fuel pool, it is usually difficult to subdivide a large complex problem into several smaller, simpler ones. To ease the computational requirements to a reasonable level, deviation from discretization schemes typically recommended by CFD code user's manuals becomes necessary.

For the recently docketed (50-382) and reviewed Waterford Unit 3 license amendment request, an exhaustive independent review of the CFD modeling of spent fuel pools was performed. This study, submitted to the NRC and approved by Amendment No. 144 for Waterford Unit 3, performed a sensitivity analysis of a Holtec CFD model which included storage of freshly discharged fuel in the Waterford cask pit. This study demonstrated that substantial computational grid refinement and the use of an alternate turbulence model Renormalization Group (RNG) had an insignificant impact on the peak local temperature result thus indicating grid independence. The independent review also recommended that a three-dimensional model would be more appropriate for evaluating fuel storage in the cask pit. The Wolf Creek CFD model is three dimensional.

The combination of the many conservative assumptions incorporated into the FLUENT analysis, the prior sensitivity studies, and the experience of Holtec's analysts in using the code for spent fuel pool analyses provides significant confidence in the results of the analysis.

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Michael J. Angus, Manager Licensing and Corrective Action at Wolf Creek Generating Station, (316) 364-8831, extension 4077.

COMMITMENT	Due Date/Event
Administrative controls will be implemented to define acceptable storage locations for spent fuel assemblies to ensure that specified radiation zone designations are not exceeded. The administrative controls will be based on calculations and will be documented in the form of plant procedures associated with special nuclear material control. The administrative controls will establish minimum cooling times for spent fuel assemblies prior to storage in designated locations.	Modification Implementation
The zone designation for the cask washdown pit area will change from "B" to "E" based on the calculated dose rates associated with the most limiting case analyzed. The zone designation above the pool surface in the cask loading pit will be changed from zone "B" to zone "C," which matches the current zone designation above the Spent Fuel Pool given on USAR Figure 12.3-2.	Modification Implementation
An underwater vacuum cleaner system will be used in conjunction with radiation surveys to ensure appropriate areas are vacuumed as part of dive preparations. During reracking, a Tri-Nuc underwater filtration unit with associated filters will be used to vacuum the Spent Fuel Pool. A backup system will also be available. Vacuuming will also consist of vacuuming crud after a series of "old" racks have been removed. Vacuuming will occur in areas in which the diver may have to enter, provided the dose rates in the area warrant vacuuming.	Modification Implementation
The use of visual barriers such as air bubbles, ropes, or orange safety netting or the use of physical barriers such as tethering the diver umbilical and other control measures will be implemented if the situation is determined to warrant it by the ALARA group or the dive controller. The diver will be in continuous voice communication with the dive controller/Health Physics personnel. In the event that an unexpectedly high radiation field/hot particle is encountered by the diver, he will be verbally instructed to take the appropriate corrective action. Positive control of the diver will be maintained, at all times, through use of an individual on the surface maintaining a diver safety line. During the planning stage, the sequence of work will be evaluated to ensure safe dive areas are	Modification Implementation

COMMITMENT	Due Date/Event
established for any diving activity. Sketches of rack layouts and positions of irradiated hardware will be reviewed by site personnel, prior to each dive, to ensure sufficient space has been provided between the diver and irradiated hardware. A pre-job brief will be conducted with the diver to discuss radiological conditions and work scope prior to each dive. Every dive situation will be evaluated to ensure the safety of the diver.	
The racks will be rinsed with demineralized water while being removed from the pool to remove contaminants to an acceptable level. Administrative controls will prevent the demineralized water from diluting the pool below acceptable boron concentration limits. After removal from the pool the racks will be bagged, sealed, down-ended, and placed into a special DOT approved shipping container. The rack will be braced inside the container, prior to sealing the container, to prevent shifting during transit. The container and enclosed rack will be shipped on a flatbed truck to Manufacturing Science Corporation (MSC) in Oak Ridge, Tennessee for disposal.	Modification Implementation
Depending upon the cycle specific fuel loading strategy and the assumptions used, the cycle specific Spent Fuel Pool heat load and pool bulk fluid temperature will be calculated to confirm that the heat load is bounded by the maximum allowable heat load of 63.41 MBtu/h, in order to ensure the calculated bulk pool temperature will remain below the established temperature limit.	First Refueling following Modification Implementation
Wolf Creek will use administrative controls to ensure that newly off-loaded fuel is not stored in the cask loading pit. The administrative controls will be documented in the form of plant procedures associated with special nuclear material control.	Upon installation of racks in the cask loading pit
A calculation of the doses for the exclusion area, low population zone, and control room to demonstrate compliance to the applicable regulatory limits will be completed prior to installation of the racks.	Prior to rack installation