FirstEnergy

Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449-9760

Guy G. Campbell
Vice President - Nuclear

419-321-8588
Fax: 419-321-8337

Docket Number 50-346

License Number NPF-3

Serial Number 2640

December 2, 2000

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Application to Increase Spent Fuel Storage Capability
(License Amendment Request No. 98-0013)

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specifications. The proposed changes involve: Technical Specification (TS) 3/4.9.7, Refueling Operations - Crane Travel - Fuel Handling Building, and associated Bases; TS 3/4.9.11, Refueling Operations - Storage Pool Water Level, and associated Bases; TS 3/4.9.12, Refueling Operations - Storage Pool Ventilation; TS 3/4.9.13, Refueling Operations - Spent Fuel Assembly Storage, and associated Bases; and TS 5.6, Design Features - Fuel Storage.

The DBNPS began operating Cycle 12 (May, 1998) with insufficient storage capacity in the spent fuel pool (SFP) to fully offload the entire reactor core (177 fuel assemblies). Since a full core offload into the SFP was required for the performance of the ten-year Inservice Inspection activities during the Spring, 2000 Twelfth Refueling Outage (12RFO), the DBNPS submitted License Amendment Request (LAR) 98-0007 (DBNPS Serial Number 2550) on May 21, 1999, to allow the use of spent fuel racks in the cask pit area adjacent to the SFP. License Amendment Number 237 was issued on February 29, 2000, providing approval for use of up to 289 cask pit rack storage locations. As described in LAR 98-0007, this added storage capability will also be utilized to provide temporary storage of fuel assemblies to support a complete re-racking of the SFP, and the four cask pit storage racks will be relocated into the SFP as part of the final completion of this re-racking project.

The purpose of this license amendment application, LAR 98-0013, is to propose the necessary revisions to the DBNPS TS to reflect an increase in SFP storage capability, as a

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result of the SFP re-racking project, from the current capacity of 735 fuel assemblies, to a new capacity of 1624 fuel assemblies. To provide additional temporary storage of fuel assemblies to support a complete re-racking of the SFP, this license amendment application also requests approval for up to 90 transfer pit storage locations. The transfer pit storage rack will be relocated into the SFP as part of the completion of this re-racking project. The resulting SFP fuel storage capacity will be sufficient to meet storage needs through the current expiration date of the DBNPS operating license, April 22, 2017.

The DBNPS requests that the enclosed license amendment application be approved by the NRC by October 1, 2001. This will support the planned commencement of the SFP re-rack modification in late-October, 2001. The SFP re-rack modification is scheduled to be complete in February, 2002, which is prior to new fuel receipt for 13RFO.

Please note that as described in the attached Affidavit (Attachment 3 to Enclosure 1), the Holtec International "Design and Licensing Report, Davis-Besse Spent Fuel Pool Rerack Project" (Attachment 4 to Enclosure 1) contains information that is considered proprietary, and, pursuant to 10 CFR 2.790, it is requested that this information be withheld from public disclosure. A non-proprietary version of the report has been prepared and is included as Attachment 5 to Enclosure 1.

Should you have any questions or require additional information, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

MKL

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
K. S. Zellers, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

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Enclosure 1
Page 1

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. The Safety Assessment and Significant Hazards Consideration is included as Attachment 1.

The proposed changes (submitted under cover letter Serial Number 2640) concern Appendix A, Technical Specifications (TS):

3/4.9.7	Refueling Operations - Crane Travel - Fuel Handling Building, and associated Bases
3/4.9.11	Refueling Operations - Storage Pool Water Level, and associated Bases
3/4.9.12	Refueling Operations - Storage Pool Ventilation
3/4.9.13	Refueling Operations - Spent Fuel Assembly Storage, and associated Bases
5.6	Design Features - Fuel Storage

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: Guy G. Campbell

Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 2nd day of December, 2000.

Nora L. Flood

Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specification (TS) 3/4.9.7, Refueling Operations - Crane Travel - Fuel Handling Building, and associated Bases; TS 3/4.9.11, Refueling Operations - Storage Pool Water Level, and associated Bases; TS 3/4.9.12, Refueling Operations - Storage Pool Ventilation; TS 3/4.9.13, Refueling Operations - Spent Fuel Assembly Storage, and associated Bases; and TS 5.6, Design Features - Fuel Storage.

A. Time Required to Implement: The License Amendment associated with this license amendment application is to be implemented within 120 days following NRC issuance.

B. Reason for Change (License Amendment Request Number 98-0013):

The purpose of this license amendment application, LAR 98-0013, is to propose the necessary revisions to the DBNPS TS to reflect an increase in spent fuel pool (SFP) storage capability, as a result of the SFP re-racking project, from the current capacity of 735 fuel assemblies, to a new capacity of 1624 fuel assemblies. To provide additional temporary storage of fuel assemblies to support a complete re-racking of the SFP, this license amendment application also requests approval for up to 90 transfer pit storage locations. The transfer pit storage rack will be relocated into the SFP as part of the completion of this re-racking project. The resulting SFP fuel storage capacity will be sufficient to meet storage needs through the current expiration date of the DBNPS operating license, April 22, 2017.

C. Attachments:

1. Safety Assessment and Significant Hazards Consideration
2. Environmental Assessment
3. Affidavit Pursuant to 10 CFR 2.790
4. "Design and Licensing Report, Davis-Besse Spent Fuel Pool Rerack Project," Holtec International, *Proprietary Version*
5. "Design and Licensing Report, Davis-Besse Spent Fuel Pool Rerack Project," Holtec International, *Non-Proprietary Version*

Docket Number 50-346
License Number NPF-3
Serial Number 2640
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 98-0013

(40 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 98-0013

TITLE:

Proposed Modifications to the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, Facility Operating License NPF-3, Appendix A Technical Specifications, to Allow an Increase in the Spent Fuel Storage Capability.

DESCRIPTION:

A facility for the long-term storage of spent nuclear fuel assemblies from commercial nuclear power reactors is to be provided by the United States Department of Energy. However, since such a facility is not yet available or expected to be available until at least the year 2010, commercial nuclear power plants, such as the DBNPS, have had to provide for additional spent fuel storage.

The DBNPS began operating Cycle 12 (May, 1998) with insufficient storage capacity in the spent fuel pool (SFP) to fully offload the entire reactor core (177 fuel assemblies). Since a full core offload into the SFP was required for the performance of the ten-year Inservice Inspection activities during the Spring, 2000 Twelfth Refueling Outage (12RFO), the DBNPS submitted License Amendment Request (LAR) 98-0007 (DBNPS Serial Number 2550) on May 21, 1999, to allow the use of four spent fuel racks in the cask pit area adjacent to the SFP. License Amendment Number 237 was issued on February 29, 2000, providing approval for use of up to 289 cask pit rack storage locations. As described in LAR 98-0007, this added storage capability will also be utilized to provide temporary storage of fuel assemblies to support a complete re-racking of the SFP, and the four cask pit storage racks will be relocated into the SFP as part of the final completion of this re-racking project.

The purpose of this license amendment application, LAR 98-0013, is to propose the necessary revisions to the DBNPS TS to reflect an increase in SFP storage capability, as a result of the SFP re-racking project, from the current capacity of 735 fuel assemblies, to a new capacity of 1624 fuel assemblies. To provide additional temporary storage of fuel assemblies being moved to support a complete re-racking of the SFP, this license amendment application also requests approval for up to 90 transfer pit storage locations (one spent fuel storage rack). This additional temporary storage would be utilized, if necessary, to minimize the dose to underwater divers needed for the SFP re-rack activities. The transfer pit is normally used to facilitate the transfer of fuel assemblies between the refueling canal in the containment vessel and the SFP. Currently, there is no storage rack in the transfer pit. The transfer pit storage rack will be relocated into the SFP as part of the completion of this re-racking project. The resulting SFP fuel storage capacity will be sufficient to meet storage needs through the current expiration date of the DBNPS operating license, April 22, 2017.

Each of the proposed revisions is shown on the attached marked-up Operating License pages. The proposed changes are described in further detail as follows:

TS 3/4.9.7 Refueling Operations - Crane Travel - Fuel Handling Building, and Associated Bases

It is proposed to revise Limiting Condition for Operation (LCO) 3.9.7, Surveillance Requirement (SR) 4.9.7, and associated Bases 3/4.9.7 to include provisions regarding storage of fuel assemblies in the transfer pit. In addition, an asterisked footnote is proposed to be added to the LCO and the SR to include an exception to the 2430-pound crane load limitation. This exception would allow an impact cover to be moved over fuel assemblies in the cask pit. This exception would also allow loads in excess of 2430 pounds to be moved over fuel assemblies in the cask pit provided: 1) an impact cover is installed, and 2) administrative controls are established to limit the load to 17,530 pounds and to limit the height that the load may travel over the impact cover. A related discussion is also proposed for addition to Bases 3/4.9.7.

TS 3/4.9.11 Refueling Operations - Storage Pool Water Level

It is proposed to revise LCO 3.9.11 and SR 4.9.11 to include provisions regarding storage of fuel assemblies in the transfer pit.

TS Bases 3/4.9.10 and 3/4.9.11 Water Level - Reactor Vessel and Storage Pool

The TS Bases presently states, in part:

> The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

The wording of this statement could be incorrectly interpreted to imply that only 10% of the total activity contained in the ruptured fuel assembly's gap is being released. The intended meaning is that 10% of the activity in the fuel assembly is in the ruptured fuel assembly's gap. As an administrative clarification, it is proposed to revise the statement to read as follows:

> The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the iodine gap activity released from the rupture of an irradiated fuel assembly.

TS 3/4.9.12 Refueling Operations - Storage Pool Ventilation

It is proposed to revise LCO 3.9.12 to include provisions allowing storage of fuel assemblies in the transfer pit.

TS 3/4.9.13 Refueling Operations - Spent Fuel Assembly Storage, and Associated Bases

It is proposed to revise LCO 3.9.13 and SR 4.9.13.1 to include provisions regarding storage of fuel assemblies in high density racks in the SFP and the transfer pit. In addition, the title of Figure 3.9-1 is proposed to be revised to clarify that it applies to the low density racks in

the SFP, and the title of Figure 3.9-2 is proposed to be revised to clarify that it applies to the high density racks in the cask pit. A new Figure 3.9-3 is proposed to be added that applies to the high density racks in the SFP and transfer pit. Related changes to Bases 3/4.9.13 are also proposed, including the addition of cross-references to Specifications 5.6.1.1 and 5.6.1.3 for a description of the design features of the low and high density spent fuel storage racks, respectively. In addition, a change to Bases 3/4.9.13 is proposed to clarify the term "directly adjacent" as used in Figure 3.9-1.

TS 5.6 Design Features - Fuel Storage

Technical Specification 5.6.1 describes the criticality design features for fuel storage. Current TS 5.6.1.1, TS 5.6.1.2, and TS 5.6.1.3 describe criticality design features specific to the SFP storage racks, the new fuel storage racks, and the cask pit storage racks, respectively. It is proposed to revise TS 5.6.1.1 to clarify that it applies to the design features of the current low density spent fuel racks located in the SFP. Similarly, it is proposed to revise TS 5.6.1.3 so that it applies to the design features of the high density spent fuel racks located in the SFP, cask pit, and transfer pit.

Technical Specification 5.6.2 describes design features related to the potential for inadvertent drainage of the water from the spent fuel storage pool and cask pit. It is proposed to revise TS 5.6.2 to include the transfer pit.

Technical Specification 5.6.3 describes the storage capacity limit of the fuel storage racks. It is proposed to revise TS 5.6.3 to include provisions for the newer high density storage racks, including storage locations in the transfer pit.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The proposed changes would allow an increase in SFP storage capability from the current capacity of 735 fuel assemblies, to a new capacity of 1624 fuel assemblies. To provide additional temporary storage of fuel assemblies to support a complete re-racking of the SFP, this license amendment application also requests approval for up to 90 transfer pit storage locations (one spent fuel storage rack). The transfer pit storage rack will be relocated into the SFP as part of the completion of this re-racking project.

These proposed changes affect the fuel handling area of the auxiliary building from a seismic/structural standpoint, as well as the spent fuel pool cooling system and decay heat removal system (when used for spent fuel pool cooling) from a thermal-hydraulics standpoint. The fuel handling area ventilation system is also affected, as are activities relating to the proper storage and handling of fuel assemblies.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

The spent fuel pool (SFP), transfer pit, and the cask pit are located within the fuel handling area of the auxiliary building, which is a reinforced concrete structure. The auxiliary building is a Seismic Class I structure which is designed to withstand seismic, tornado, and

thermal loads, as discussed in DBNPS Updated Safety Analysis Report (USAR) Sections 3.7, "Seismic Design," and 3.8, "Design of Seismic Class I and Class II Structures." The spent fuel storage racks are also Seismic Class I structures which are designed to withstand seismic loadings.

As shown in Figure 3.5.1 of Reference 4 (attached), the transfer pit and cask pit are along the entire west wall of the SFP. The transfer pit is north of the cask pit. The three areas are separated from each other by 3-foot-thick concrete walls. The only communication between the SFP and the cask pit is through a 36-inch-wide slot opening. This opening is provided with a watertight bulkhead that can isolate the SFP when needed. Similarly, the only communication between the SFP and the transfer pit is through a 36-inch-wide slot opening, which is also provided with a watertight bulkhead that can isolate the SFP when needed. There is no communication between the transfer pit and the cask pit. The floor of the cask pit is approximately 6.5 feet lower than the floors of the SFP and the transfer pit.

The functions of the SFP are to support the SFP racks and retain the SFP coolant during normal operations and abnormal conditions. Spent fuel storage is described in USAR Section 9.1.2, "Spent Fuel Storage." The SFP is a reinforced-concrete pool lined with stainless steel. The pool is currently sized to store 720 irradiated fuel assemblies. In addition, the pool currently contains storage locations for 15 failed fuel containers, for a total of 735 storage locations. The spent fuel storage cells are installed in parallel rows with center-to-center spacing of 12-31/32 inches in one direction, and 13-3/16 inches in the other direction. Each cell consists of a 9-inch square stainless steel can. The water gap between the stainless steel cans produces what is known as a "flux trap." The "flux trap" construction is sufficient to maintain a k_{eff} of 0.95 or less for spent fuel of initial enrichment of 3.56 wt% U-235 or less, assuming the storage racks are flooded with unborated water. Higher enrichment spent fuel assemblies must be stored in a checkerboard pattern, taking into account fuel burnup, to maintain a k_{eff} of 0.95 or less. A hole in the bottom of each spent fuel storage cell allows coolant to flow up through the seated fuel assemblies.

The cask pit is designed to provide for the transfer of the spent fuel assemblies from storage to a shipping cask or dry fuel storage canister. In addition, up to 289 fuel assemblies are authorized to be stored in four cask pit racks. However, the cask pit racks will be relocated into the SFP as part of the completion of the planned SFP re-racking project.

The transfer pit currently provides for the transfer of fuel assemblies between the refueling canal in the containment vessel and the SFP, via the fuel transfer tubes, which are sealed closed during plant operation.

All spent fuel assembly transfer operations are normally conducted under a minimum of 9-1/2 feet of borated water above the top of the active fuel to ensure adequate biological shielding. The SFP, cask pit, and transfer pit are protected against inadvertent draining. All penetrations in the SFP, cask pit, and transfer pit are more than 9 feet above the top of fuel assemblies stored in the racks. The spent fuel pool cooling discharge piping continues down to near the bottom of the SFP, after entering the pool at the same elevation as the suction. The discharge piping includes a half-inch diameter anti-siphon hole. Drain lines from the SFP, cask pit, and transfer pit are isolated via locked closed valves. Operation of locked

valves is subject to administrative controls. In addition, low SFP level and radiation monitor alarms in the control room provide early warning in the event of a loss of SFP inventory.

The SFP water is cooled by the spent fuel pool cooling system, as discussed in USAR Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." The spent fuel pool cooling system is shown in USAR Figure 9.1-5. It consists of two half system capacity recirculating pumps, two half system capacity heat exchangers, and associated valves, piping and instruments. The spent fuel pool cooling system is currently designed to maintain the borated SFP water at 125 °F or less with a heat load of 12.4×10^6 Btu/hr. If it becomes necessary to off-load an entire core into the SFP, cooling can be provided by the decay heat removal system, which has a higher heat removal capacity. The decay heat removal system (DHRS), which is described in USAR Section 6.3, "Emergency Core Cooling System," also serves as a Seismic Class I backup system to the spent fuel pool cooling system.

The current operating philosophy for SFP cooling, as reflected in the DBNPS shutdown risk program, is based on the core offload scenario. With the core not fully offloaded, one SFP cooling train should be functional if a DHR train is available for SFP cooling. Both SFP cooling trains should be functional if neither DHR train is available for SFP cooling. With the core fully offloaded, both SFP cooling trains should be functional and one DHR train should be functional for SFP cooling. The DHR train may be temporarily removed from functional status to support other outage evolutions, provided specific provisions are enacted to ensure that the DHR train remains readily available to support SFP cooling.

The spent fuel pool cooling system components are operated to maintain the pool temperature less than 125 °F. The component cooling water (CCW) system provides cooling to the SFP heat exchangers. The combinations of pumps and heat exchangers utilized are dependent on the component cooling water system temperature.

There are alarms provided for the spent fuel pool cooling system to indicate high or low SFP water level. The low level alarm assures a minimum of 23 feet of water is maintained above the fuel assemblies. The high level alarm is provided to prevent overfill. There is an overfill line which limits the maximum level to 601 feet 9 inches. The SFP water is maintained at a normal level of Elevation 601 feet 6 inches. The setpoints of the high and low level alarms are Elevations 601 feet 7 inches and 601 feet 2 inches, respectively.

A SFP water temperature indicator is provided in the control room. Spent fuel pool temperature indication is also available via the plant computer. A SFP high temperature annunciator alarm is also provided in the control room, with a setpoint of 125 °F.

In addition to its primary function, the spent fuel pool cooling and cleanup system provides purification by removing fission and corrosion products in the SFP, cask pit, and transfer pit water. It can also be aligned to purify the contents of the borated water storage tank. The SFP pumps take suction from the SFP and recirculate the water back to the pool after it passes through the SFP heat exchangers, and the demineralizer and/or filter in various combinations, as required.

The fuel handling area ventilation system is described in USAR Section 9.4.2.2, "Fuel-Handling Area." The system is designed to provide an average of 20 air changes per hour over the surface of the SFP, and to maintain the fuel handling area at between 60 and 110 °F. The ventilation flow for the fuel handling and storage area housing the SFP, the cask pit, and transfer pit is normally exhausted to the environment through the station vent stack. Exhaust air from the fuel-handling area is monitored by radiation detectors before it is discharged from the station through the vent stack. Upon detection of a fuel handling accident, the emergency ventilation system (EVS) is automatically started and charcoal iodine adsorber filters are utilized to filter exhaust air from the fuel handling area, via interconnections.

The spent fuel cask crane is comprised of a main hook rated for 140 tons, as well as an auxiliary hook rated for 20 tons. The design of the spent fuel cask crane prevents it from traveling over the SFP, cask pit, and transfer pit unless a key-operated bypass switch is actuated.

EFFECTS ON SAFETY:

The attached "Design and Licensing Report, Davis-Besse Spent Fuel Pool Rerack Project," Holtec International (Reference 4), provides the technical basis for the proposed changes to the Technical Specifications.

Summary of Technical Evaluation

General

The current spent fuel storage capability includes 720 storage locations in the spent fuel pool (SFP) and 289 storage locations in the cask pit. In addition, there are 15 storage locations in the SFP for failed fuel containers, which have never been used. As part of the plant modification to re-rack the SFP, all of the low density rack modules in the current SFP will be replaced with high density rack modules, containing 1624 total storage locations. There will be no dedicated storage locations in the new rack modules for failed fuel containers. In the event a damaged fuel assembly could not be repaired, a failed fuel container would be designed to fit into a new high density rack module, or else a holder would be designed to store a larger failed fuel container upright in the cask pit or other suitable location. The four cask pit high density rack modules (289 total storage locations) currently approved for use will be utilized to provide temporary storage of fuel assemblies during the re-racking, however they will be eventually be emptied and relocated to the SFP as part of the modification.

Underwater divers will be needed for the SFP re-rack activities. In addition to the four cask pit modules, if necessary, additional temporary storage of fuel assemblies is planned to minimize the dose to the underwater divers. The technical evaluation considers the temporary placement of one of the new high density SFP rack modules (up to 90 storage locations) in the transfer pit. Like the cask pit rack modules, the transfer pit rack module will eventually be emptied and relocated to the SFP as part of the modification. Placement of a rack module in the transfer pit will interfere with the ability to transfer fuel assemblies between the refueling canal in the containment vessel

and the SFP. In the event that it becomes necessary to transfer fuel assemblies, the fuel assemblies in the transfer pit rack module would first need to be transferred to the SFP and the rack module removed from the transfer pit. The re-racking process maintains sufficient room within the SFP to transfer these fuel assemblies to the SFP.

The design of the new SFP rack modules is identical to the design of the rack modules that were approved for use in the cask pit. The new SFP rack modules are freestanding and self-supporting. The principle construction materials for the racks are ASME SA240-Type 304 stainless steel and plate stock, and ASME SA564-630 precipitation-hardened stainless steel for the adjustable pedestals. The only non-stainless material utilized in the rack is the neutron absorber material, which is a hot-rolled cermet of boron carbide and aluminum, clad in aluminum (patented product name "Boral™"). Boral is chemically inert and has been used extensively in nuclear unit SFP environments. Tables 3.3.1 and 3.3.2 of Reference 4 (attached) provide a listing of nuclear units that have previously used Boral in spent fuel storage rack applications.

Additional details regarding the design and construction of the new SFP racks are provided in Reference 4 (attached). Section 1.0 of Reference 4 provides a brief introduction and includes a diagram of the proposed SFP layout (Figure 1.1). Section 2.0 of Reference 4 provides an overview of the new racks, including a detailed description of the rack module geometry and construction. As described in Section 3.0 of Reference 4, all materials used in the construction of the new racks are compatible with the SFP environment.

Criticality Safety Evaluation

Section 4.0 of Reference 4 (attached) provides details on the criticality safety evaluation.

The criticality analyses qualify the high density rack modules for storage of fuel assemblies in one of three different loading patterns, subject to certain restrictions: Mixed Zone Three Region, Checkerboard, and Homogeneous Loading. The proposed new TS Figure 3.9-3 provides the Category-specific burnup/enrichment limitations. Based on the above limitations, different loading patterns may be used in different rack modules, provided each rack module contains only one loading pattern. With additional restrictions, two different loading patterns may be used in a single rack module. The loading pattern restrictions will be maintained in fuel handling administrative procedures (in a manner similar to that approved by the NRC for the Callaway Plant, Unit 1 -- Reference 13).

The criticality analyses show that the maximum neutron multiplication factor, k_{eff}, is less than or equal to 0.95, including uncertainties (e. g., water density, calculational uncertainties, and manufacturing tolerances), for all normal and accident conditions.

The criticality analyses utilize conservative assumptions. The SFP water was assumed to be at a temperature that results in the highest reactivity. In addition, no soluble poison (boron) was assumed present in the water under normal operating conditions. However, as described in Section 4.6.3 of Reference 4, in the event that a unirradiated

fuel assembly of highest permissible enrichment were inadvertently misloaded into a rack location intended for burned fuel or into an empty rack location between other fresh assemblies intended to be stored in a checkerboard pattern, or accidentally mislocated outside of a storage rack adjacent to other fuel assemblies, credit is taken for soluble boron in the water to ensure that k_{eff} remains less than or equal to 0.95. Administrative controls have been established to ensure that the SFP boron concentration is maintained at ≥ 1800 ppm during and following fuel movement, until completion of verification that no misloading has occurred.

The effects on criticality due to a dropped fuel assembly that falls across the top of already stored fuel are described in Section 4.6.4 of Reference 4. The active fuel of the dropped fuel assembly remains more than 12 inches away from the active fuel in the storage rack, therefore the effect on reactivity will be insignificant and the configuration is assured to remain subcritical.

The loading pattern restrictions for the SFP rack modules can be conservatively applied to the new rack module in the transfer pit. The restrictions are conservative due to the fact that a single rack module in the transfer pit would have an increased neutron leakage and hence a lower reactivity, compared to the multiple rack module configuration in the SFP.

Thermal-Hydraulics Evaluation - Spent Fuel Pool

A comprehensive thermal-hydraulic evaluation was performed in support of the license amendment application that was submitted to allow storage of fuel assemblies in the cask pit racks (Reference 8). NRC review of this evaluation is documented in the safety evaluation associated with License Amendment No. 237 (Reference 9). Since the thermal-hydraulic evaluation assumed a re-racked SFP loaded to maximum capacity, that evaluation is bounding. The details of the evaluation are provided in Section 5.0 of Reference 4 (attached).

Based on a conservative evaluation of the projected spent fuel discharge schedule for the DBNPS, the analyses determined the maximum bulk and local temperatures that would result from a worst case SFP heat load of 30.15 x 10^6 BTU/hr. This heat load value is based on the SFP filled to capacity. Consistent with the evaluation, the maximum total heat generation rate of a single fuel assembly stored in the SFP is limited to 80,209 watts (273,870 BTU/hr). In addition, the maximum heat generation rate per heat transfer surface area of assembly cladding is limited to 445 watts/ft^2 (1520 BTU/hr-ft^2). These limits will be included in the USAR Technical Requirements Manual (TRM). Future changes to the USAR TRM will be evaluated under the requirements of 10 CFR 50.59, and the NRC will be informed of these changes in accordance with the USAR update requirements of 10 CFR 50.71(e).

The thermal-hydraulic analyses utilized USAR-specified capabilities of the spent fuel pool cooling system and its backup, the decay heat removal system. The minimum time-to-boil and maximum boil-off rate were determined based on a loss of SFP forced cooling with the maximum initial water temperature and the corresponding heat load.

After the completion of the thermal-hydraulic analyses for the re-racked SFP, the DBNPS received License Amendment Number 242 in September, 2000 (Reference 11), which increased the maximum allowable Ultimate Heat Sink (UHS) temperature from 85 °F to 90 °F. This change affects the Component Cooling Water (CCW) heat exchanger outlet temperature assumed in the thermal-hydraulic analyses performed for the SFP re-rack. The DH Coolers and SFP Coolers provide cooling for the SFP and are cooled by CCW. The CCW heat exchangers are cooled by the Service Water (SW) system, which draws suction from the UHS. The impact of the CCW temperature increase on the thermal-hydraulic analyses was evaluated and is described in a later section.

The worst-case heat load that must be rejected from the SFP will occur when fuel from the reactor is discharged to the SFP. Four discharge scenarios were analyzed:

Scenario 1 considered a partial core discharge of 72 fuel assemblies from the reactor into a SFP that already contains 1609 previously discharged fuel assemblies with a minimum decay time of two years, for a total of 1681 stored fuel assemblies. The projected discharge schedule that would yield this cumulative number of stored fuel assemblies is shown in Table 5.8.3 of Reference 4 (attached). This analyzed spent fuel inventory exceeds the maximum SFP re-rack inventory of 1624 fuel assemblies, but was used to provide a clearly conservative thermal loading. Thus, the results from the analyzed partial core discharge scenarios of 1681 stored assemblies bound the maximum inventory of 1624 stored assemblies. Two SFP pumps and two SFP heat exchangers were assumed to be operating.

Scenario 2 is similar to scenario 1, except that only one SFP pump and heat exchanger were assumed to be operating.

Scenario 3 considered a full core discharge of 177 fuel assemblies from the reactor into a SFP that already contains 1537 previously discharged fuel assemblies, for a total of 1714 stored fuel assemblies. The projected discharge schedule that would yield this cumulative number of stored fuel assemblies is shown in Tables 5.8.4 and 5.8.5 of Reference 4. This analyzed spent fuel inventory exceeds the maximum SFP re-rack inventory of 1624 fuel assemblies, but was used to provide a clearly conservative thermal loading. Thus, the results from the analyzed partial core discharge scenarios of 1714 stored assemblies bound the maximum expected inventory of 1624 stored assemblies. Regarding the previously discharged fuel assemblies, two cases were run. Scenario 3A considered a minimum decay time of 65 days for the most recent batch of previously discharged fuel, i.e., a 65 day decay of the fuel in the SFP after the planned refueling outage (coincident with 65 days of operation of the refueled reactor core), then an unplanned shutdown (i.e., "second reactor shutdown") leading to a full core discharge. Scenario 3B considered a minimum decay time of two years (coincident with two years of operation of the refueled reactor core) leading to a full core discharge. Two SFP pumps and two SFP heat exchangers were assumed to be operating.

Scenario 4 is similar to scenario 3, including the two cases, except that cooling is provided by the decay heat removal system.

Cooling system alignments for scenarios 2 and 3 would not typically be used during fuel discharge operations under the most adverse conditions, and, therefore, results are only compared to the bulk boiling temperature of 212 °F. For a partial core discharge, two spent fuel pumps and heat exchangers would normally be available. For a full core discharge, the decay heat removal system is available for SFP cooling, as conditions warrant. Scenarios 2 and 3 were included to demonstrate that in the event of a spent fuel pool cooling system malfunction, the bulk temperature remains below boiling for these scenarios. For scenarios 1 and 4, the acceptance criterion used for the analysis is that the pool bulk temperature remain within the limits of the American Concrete Institute (ACI) "Code Requirements for Nuclear Safety Related Concrete Structures," (Reference 6) to protect the integrity of the structure. The ACI Code permits long-term temperatures of up to 150 °F and short-term temperature excursions in localized areas up to 350 °F.

For scenario 1, the peak bulk pool temperature was determined to be approximately 133 °F, which meets the long-term acceptance criterion of 150 °F. For scenarios 2, 3A, and 3B, the peak bulk pool temperatures were determined to be approximately 169 °F, 166 °F, and 165 °F, respectively. These temperatures are substantially below the boiling point, thereby meeting the analysis acceptance criterion. For scenarios 4A and 4B, the peak bulk pool temperatures were determined to be approximately 151.5 °F and 150.7 °F, respectively. Although these bulk temperatures are slightly above the long-term limit of 150 °F, the time for which the limit will be exceeded is less than 28 hours for both scenarios. Since the ACI Code allows short-term temperature excursions as high as 350 °F in localized areas, the calculated results are acceptable.

The evaluation of the effects of a complete failure of the forced cooling systems, which is assumed to occur with the SFP bulk temperature at a maximum, shows there would be at least 10.42 hours available, prior to the beginning of bulk boiling, for corrective actions to restore cooling for the partial core discharge scenario (scenario 1), and at least 3.78 hours for the most limiting full core discharge scenario (scenario 4A). As previously noted, scenarios 2 and 3 were not included in this evaluation. The evaluation also shows that the maximum boil-off rate, should corrective actions not be successful, would be less than 35 gpm for scenario 1, and less than 70 gpm for scenario 4A. Low level and high temperature alarms are provided for the SFP. The minimum time of 3.78 hours to reach bulk boiling conditions in the SFP following a loss of all forced cooling is comparable to the time calculated for similar analyses in support of licensing actions for other dockets (see Reference 5, for examples).

In the unlikely event that a boil-off situation were to occur, in order to maintain SFP water level, make-up water can be provided via a variety of already-proceduralized valve line-ups, including gravity fill methods. Make-up to the SFP can be provided from the borated water storage tank via the decay heat removal system, as shown in USAR Figure 9.1-6, "Spent Fuel Pool Make Up Water From Seismic Class I System." The available makeup rate from this source exceeds the maximum 70 gpm boil-off rate.

Make-up to the SFP is also available from the Seismic Class II demineralized water storage tank and clean waste receiver tanks. The available makeup rate from each of these sources also exceeds the 70 gpm boil-off rate.

In the unlikely event that the establishment of makeup to the SFP was delayed during a boil-off event, approximately 25 hours would be required to reduce SFP level from the TS minimum level of 23 feet above the top of fuel assemblies seated in the storage racks, to the level corresponding to 9-1/2 feet above the top of fuel stored in the racks, given a SFP plan area of approximately 1057 ft^2, and assuming a constant boil-off rate of 70 gpm. This conservatively assumes no credit for the volume in the cask pit or transfer pit. A minimum of 9-1/2 feet of borated water above the top of active fuel stored in the racks will ensure adequate biological shielding. This 25-hour period provides operators with more than sufficient time to intervene with available means to maintain or restore the SFP water level.

The maximum SFP local water temperature and maximum fuel cladding temperature were also determined, considering a full core discharge into the SFP, with decay heat from 1714 fuel assemblies (as assumed in the analyses for Scenarios 3 and 4). Thus, as with the bulk temperature analyses previously described, the results from the analyzed scenario of 1714 stored assemblies bound the maximum SFP re-rack inventory of 1624 stored assemblies. The maximum SFP local water temperature and maximum fuel cladding temperature were calculated to be approximately 194 °F and 230 °F, respectively. Considering the pressure due to the depth of water (23 feet), the saturation temperature at the top of the spent fuel storage racks is approximately 239 °F. Therefore, the above results confirm that local boiling will not occur in the SFP.

An abnormal SFP temperature would be detected by routine monitoring of control room indication. The indicator reading is logged by the control room operator once per 8 hours. The log alerts the operator that additional attention is warranted should the SFP water temperature reach 120 °F. Spent fuel pool temperature indication is also available via the plant computer. In addition, a SFP high temperature annunciator alarm is provided in the control room, with a setpoint of 125 °F. Upon receipt of the alarm, the alarm procedure directs the operator to: check for SFP high temperature by observing the control room SFP temperature indicator or computer point; check that the SFP heat exchanger outlet temperatures are less than 100 °F; verify adequate component cooling water (CCW) flow rate to each SFP heat exchanger if the SFP heat exchanger outlet temperature is greater than 100 °F; take appropriate actions if CCW flow rate is not adequate; and raise cooling capacity by starting a second SFP pump if only one SFP pump is running. If the SFP cooling system has been lost or is insufficient to maintain SFP water temperature below 125 °F, the alarm procedure directs the operator to utilize the DHRS.

If a DHR train being utilized for SFP cooling is lost, and no DHR train can be aligned to provide SFP cooling, the abnormal procedure instructs the operator to place both trains of SFP cooling in service, if available. In the event the SFP temperature reaches 125 °F, the procedure further directs the operator to evacuate the SFP area and place the

Emergency Ventilation System in service on the SFP area.

Thermal-Hydraulics Evaluation - Transfer Pit

As described in the previous section, a comprehensive thermal-hydraulic evaluation was performed in support of the license amendment application that was submitted to allow storage of fuel assemblies in the cask pit racks. Since the evaluation assumed a re-racked SFP loaded to maximum capacity, that evaluation is bounding for the SFP. The previous evaluation, however, did not address the effects of the temporary placement of one of the new SFP rack modules in the transfer pit. The details of the additional evaluation performed for the transfer pit are provided in Section 5.10 of Reference 4 (attached).

The cooling mechanism for the transfer pit is similar to the cask pit in that it is connected to the SFP by a three-foot wide gate. In addition, like the cask pit, the transfer pit has no forced cooling. The thermal-hydraulic analysis completed for the cask pit (Reference 8) concluded there is adequate buoyancy driven flow through the gate to appropriately cool 289 fuel assemblies having a total heat output of 252,200 watts. Similar results would be expected for the transfer pit, however, the maximum heat load of the transfer pit was very conservatively determined assuming a closed gate, with only passive heat loses to the building environment off the water surface. For this analysis, a building ambient temperature of 110 °F and a relative humidity of 100% were assumed. The maximum heat load was calculated with the bulk temperature conservatively limited to 140 °F. This bulk temperature restriction ensures that the American Concrete Institute (ACI) "Code Requirements for Nuclear Safety Related Concrete Structures," (ACI-349) (Reference 6) long term limit of 150 °F for concrete structures is not exceeded, and ensures that there will be no bulk boiling. The maximum heat load was determined to be 88,110 watts (300,806 BTU/hr).

The analysis also determined the evaporation rate to be 0.542 gpm. At this rate, it would take greater than 5 days to lower the level of the transfer pit from the normal operating level to the TS 3/4.9.11 minimum required level of 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. Borated water may be added to the transfer pit from the Borated Water Storage Tank (BWST), either by gravity fill or via the BWST Transfer Pump. Therefore, it can be concluded that sufficient time for remedial actions is available, and that makeup capacity will exceed the makeup demand.

Given the maximum bulk temperature of 140 °F, and adding the local temperature differences calculated for the hottest location in the SFP, the maximum local water temperature in the transfer pit is calculated to be 183 °F, and the maximum fuel cladding temperature is calculated to be 219 °F. These temperatures are less than the 239 °F boiling temperature at the top of the rack. Therefore, it can be concluded that boiling will not take place anywhere in the transfer pit with the gate installed.

In addition to the closed gate analysis, an evaluation of the fuel transfer pit cooling with the gate open was performed. A completely re-racked and filled SFP was assumed. It

was also assumed that the transfer pit bulk temperature would reach equilibrium at 4 °F above the SFP bulk temperature. This 4 °F temperature increase is conservative in that it was calculated for the cask pit, which has a maximum heat load of 252,200 watts, as compared to the transfer pit maximum heat load of 88,110 watts. It was also conservatively assumed that the bulk-to-local water temperature difference and the local water-to-fuel cladding temperature difference are the same as determined for the hottest assembly in the SFP.

The same four discharge scenarios analyzed for the SFP were considered. The same acceptance criteria are also applicable. For scenarios 1 and 4, the acceptance criterion is that the pool bulk temperature remains within the limits of ACI-349 (Reference 6) to protect the integrity of the concrete structure. The ACI Code permits long-term temperatures of up to 150 °F and short-term temperature excursions in localized areas up to 350 °F. For the reasons previously described, scenarios 2 and 3 were included to demonstrate that in the event of a spent fuel pool cooling system malfunction, the bulk temperature remains below boiling for these scenarios.

For scenario 1, the peak bulk temperature in the transfer pit with the gate open was determined to be approximately 137 °F. For scenarios 2, 3A, and 3B, the peak bulk temperatures were determined to be approximately 173 °F, 170 °F, and 169 °F, respectively. These temperatures are substantially below the boiling point, thereby meeting the analysis acceptance criterion. For scenarios 4A and 4B, the peak bulk pool temperatures were determined to be 155.4 °F and 154.7 °F, respectively. Although these bulk temperatures are above the long-term limit of 150 °F, the time for which the limit will be exceeded is approximately 60 hours for both scenarios. Since the ACI Code allows short-term temperature excursions as high as 350 °F, the calculated results are acceptable. Therefore, it can be concluded both the transfer pit bulk water and the transfer pit structure temperatures will be maintained at acceptable levels with the transfer pit-to-SFP gate open.

With the gate open and the extra volume of the transfer pit available, in the event of a complete failure of the SFP forced cooling systems, the calculated time-to-boil and boil-off rate will be bounded by the SFP analyses.

Given the maximum bulk temperature of 155.4 °F (scenario 4A), and adding the local temperature differences calculated for the hottest location in the SFP, the maximum local water temperature in the transfer pit is calculated to be approximately 198 °F, and the maximum fuel cladding temperature is calculated to be approximately 234 °F. These temperatures are less than the 239 °F boiling temperature at the top of the rack. Therefore, it can be concluded that boiling will not take place anywhere in the transfer pit with the gate open.

In conclusion, fuel may be stored in the transfer pit with the transfer pit-to-SFP gate either closed or open. The analysis limits the transfer pit total heat load to 88,110 watts. This limit will be included in the USAR Technical Requirements Manual (TRM). Future changes to the USAR TRM will be evaluated under the requirements of

10 CFR 50.59, and the NRC will be informed of these changes in accordance with the USAR update requirements of 10 CFR 50.71(e).

Impact of CCW Temperature Increase on Thermal-Hydraulic Evaluation

As previously mentioned, the above-referenced thermal-hydraulic evaluation was performed in support of the license amendment application that was submitted in May, 1999 to allow storage of fuel assemblies in the cask pit racks (Reference 8). The thermal-hydraulic evaluation assumed a maximum Component Cooling Water (CCW) heat exchanger outlet temperature of 95 °F. The DH Coolers and SFP Coolers are cooled by CCW and provide cooling for the SFP. The CCW heat exchangers are cooled by the Service Water (SW) system, which draws suction from the Ultimate Heat Sink (UHS).

After the completion of the thermal-hydraulic evaluation, the DBNPS submitted a license amendment application in July, 1999 that proposed an increase in the maximum allowable UHS temperature from 85 °F to 90 °F (Reference 10). The evaluation of the CCW heat exchangers in support of this license amendment application determined that a CCW heat exchanger outlet temperature of 97 °F could be maintained with the increased UHS temperature. This license amendment application was approved by the NRC via issuance of License Amendment No. 242 on September 12, 2000 (Reference 11).

As summarized above, under the worst-case transient conditions, the bulk SFP water temperature will be above 150 °F for less than 28 hours, and should not exceed 151.5 °F, and the bulk transfer pit temperature (with the gate open) will be above 150 °F for approximately 60 hours, and should not exceed 155.4 °F. An increase in CCW heat exchanger outlet temperature of 2 °F would increase the time above 150 °F to approximately 80 hours for the SFP and approximately 135 hours for the transfer pit. The maximum expected bulk temperatures would be less than 154 °F for the SFP and less than 158 °F for the transfer pit. These temperatures and durations are not significant when considering the thermal inertia of the concrete walls and floor. The concrete temperature will lag the water temperature such that the bulk of the concrete mass cross-section will remain well below the 150 °F range. In fact, a very small depth of concrete will actually experience temperatures in excess of 150 °F. Accordingly, the SFP and fuel transfer pit wall and floor concrete temperatures in excess of 150 °F, under the worst case conditions, are acceptable, since they will exist for a short duration. This evaluation is based on information obtained from Reference 12.

For every degree the maximum bulk temperature of the SFP or transfer pit increases due to the increase in CCW temperature, the maximum local water temperature and the maximum fuel cladding temperature will increase by approximately the same amount. Therefore, with the 2 °F increase in CCW temperature, the SFP maximum local temperature will increase from 194 °F to approximately 197 °F, the SFP maximum cladding temperature will increase from 230 °F to approximately 233 °F, the transfer pit maximum local temperature will increase from 198 °F to approximately 201 °F, and the transfer pit maximum cladding temperature will increase from 234 °F to

approximately 237 °F. As these temperatures do not exceed the saturation temperature at the top of the racks (239 °F), the acceptance criteria are satisfied.

With the SFP maximum bulk temperature increased due to the change in CCW temperature, the time-to-boil for Scenario 1 will decrease from 10.42 hours by approximately 20 minutes, and the time-to-boil for Scenario 4A will decrease from 3.78 hours by approximately 10 minutes. These small reductions in time still allow sufficient time to establish makeup as discussed earlier. Due to the increased water volume and surface area, the time-to-boil for both scenarios would be increased with the transfer pit-to-SFP gate open. With the gate closed, the heat load allowed in the transfer pit is not sufficient to cause boiling. The boil-off rate would be unaffected by the change in the maximum bulk temperature as the heat load would not change significantly with the 10 to 20 minute change in the time to reach boiling.

Structural and Seismic Evaluation

Sections 6.0 and 8.0 of Reference 4 (attached) provide details on the structural evaluation relative to the use of the new spent fuel storage racks in all normal, seismic, and accident conditions. The evaluation considered the loads from seismic, thermal, hydraulic, and mechanical forces to determine the margin of safety in the structural integrity of the fuel racks, and the spent fuel storage pool structure.

Storage Rack Evaluation

The seismic analysis was performed using the vendor's "Whole Pool Multi-Rack" analysis methodology. The analysis was based on simulations of the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE). The rack modules were analyzed as completely full, partially full and nearly empty. The fuel weight conservatively includes the additional mass of control elements considered stored integrally with every assembly.

The results indicate that the maximum seismic displacements do not result in any impacts with the pool walls or between the tops of the storage racks. Some impact forces are predicted between the baseplates of adjacent racks, but this is expected, since the racks are modeled as initially touching along the entire length of the baseplate. The resultant member and weld stresses in the racks are all below the allowable stresses. Therefore, the racks will remain functional during and after SSE and OBE events.

The predicted dynamic behavior and corresponding displacements and stresses of the single rack module to be temporarily located in the transfer pit are bounded by the analyses described above.

The rack analysis provides pedestal-to-bearing-pad impact loads resulting from lift-off and subsequent resettling during dynamic events. The pool floor stresses were evaluated for these impact loads and determined to remain within allowable limits even when considering the worst case pedestal location with respect to leak chases.

The existing spent fuel rack modules are not freestanding. There are seismic braces which extend out from the racks to the SFP walls (but are not attached to the walls). These braces may be removed in order to disassemble and remove the existing racks. Dynamic analyses were performed for the existing racks to establish kinematic stability in the unlikely scenario that a seismic event was to occur during the re-racking process. The analyses demonstrate that the existing racks are kinematically stable in the interim re-racking configuration.

In addition to the seismic evaluations, the new storage racks were also analyzed with respect to impact loads due to the accidental drop of a fuel assembly. This analysis is discussed in a separate section below.

In case of a stuck fuel assembly in the rack, an evaluation of the rack's ability to withstand a 500-pound force was performed. The resultant load on the cell walls and welds will not affect the rack structural integrity. For a 500 pound load applied vertically along a cell wall, the resultant stress is well below the yield stress of the material. For a 500 pound load applied at a 45-degree angle to the top of a cell wall, minor tear-out at the top of the cell wall, well above the top edge of the neutron absorber material will only occur.

Spent Fuel Pool and Fuel Handling Building Structural Evaluation

The Auxiliary Building consists of cast-in-place, monolithic, reinforced concrete interior and exterior walls, and is designed as a seismic Class I structure. The SFP and transfer pit represent a portion of the overall structure and are cast-in-place, steel-lined, reinforced concrete pits. There are four removable steel struts located in the transfer pit, which span between the SFP west wall and the transfer pit west wall. These struts are designed to support the wall between the transfer pit and the SFP, and must be installed prior to transfer pit drainage below the depth of the bottom of the gate.

The walls were analyzed using individual loads and load combinations in accordance with the DBNPS USAR, and based on the "ultimate strength" design method. The primary loads considered were:

- Dead weight of the concrete structure, fully loaded racks, and the water,

- Seismic OBE and SSE loads:
  vertical rack loading
  water mass and sloshing loads
  hydrodynamic pressure due to rack motion
  inertial forces of the structure

- Thermal loads producing the largest temperature gradient across the thickness of the walls.

In addition to the loads described above, the structure and liner were also analyzed for mechanical loads under accident conditions. Analyses were also performed to ensure liner integrity. The result of the analyses performed indicate that under all postulated loadings, the floor slabs, walls, and liner will be subjected to stresses or strains within acceptable limits.

Fuel Handling Accidents (USAR Chapter 15)

Spent fuel assemblies are handled entirely under water. As described in USAR Section 15.4.7, "Fuel Handling Accident," mechanical damage to the fuel assemblies during fuel handling operations is possible, but improbable. An evaluation of the consequences of a fuel handling accident outside containment is provided in USAR Section 15.4.7.2, "Accident Analysis – Accident Outside Containment." The evaluation assumes that the entire outer rows of fuel rods (56 of 208 rods), in a fuel assembly that has undergone 72 hours of decay time, suffers mechanical damage to the cladding. It is also important to note that the consequences of a fuel handling accident outside containment are bounded by the consequences of a fuel handling accident inside containment, which assumes that all 208 rods in a fuel assembly are damaged. An evaluation of the consequences of a fuel handling accident inside containment is provided in USAR Section 15.4.7.3, "Accident Analysis – Accident Inside Containment." This analysis assumes that all rods in a fuel assembly that has undergone 72 hours of decay time suffer mechanical damage to the cladding.

The new racks do not change the height of the stored fuel relative to any load being handled, and the 72 hour decay time is conservative. Therefore, the design bases fuel handling accident for the pool area remains unchanged.

Rack Structural Performance (Impact Loads)

The rack structural performance has also been analyzed with respect to impact loads due to the accidental drop of a fuel assembly during movement to a storage location. The details of the evaluations are provided in Section 7.0 of Reference 4 (attached).

In the evaluation of fuel handling accidents discussed herein, the concern is with the damage to the storage racks, and the structure. The configuration of the rack cell size, spacing, and neutron absorber material must remain consistent with the configurations used in the criticality and thermal-hydraulic evaluations. Maintaining these design configurations will ensure that the results of the criticality and thermal-hydraulic evaluations remain valid.

Two categories of fuel assembly drop accidents were evaluated, a "shallow drop" and a "deep drop":

Shallow Drop

This evaluation considers a fuel assembly, an inserted control element assembly, and the portion of the handling tool which is severable in the event of a single element failure (inner mast), dropping vertically from the highest elevation that

the load can be lifted. The criticality evaluation limits the gross cell wall deformation of the impacted and eight surrounding cells to 8.75 inches. The thermal-hydraulic evaluation for the racks assumed a flow blockage of 50% after a drop accident.

The first scenario considers the load striking the top of a stored fuel assembly and subsequently impacting the top of the rack module (note that the top of a stored fuel assembly extends above the top of the rack module). The results of the evaluation show that the top of the impacted region undergoes localized deformation, however the maximum gross deformation is limited to 3 inches of penetration. In addition, approximately 10% of the opening of the impacted cell is blocked. These results meet the above-mentioned acceptance criteria.

To maximize cell wall deformation, the second scenario considers the load striking the top of an empty rack cell. The results of the evaluation show that local damage to the impacted region of the rack is significantly more extensive than for the first scenario, however, the effective damaged area of the impacted cell measures 5 inches deep and blocks less than 50% of the cross-sectional area. These results meet the above-mentioned acceptance criteria.

<u>Deep Drop</u>

This evaluation also considers a fuel assembly, an inserted control element assembly, and the portion of the handling tool which is severable in the event of a single element failure (inner mast), dropping vertically from the highest elevation that the load can be lifted.

The first scenario considers the load dropping through an empty cell located above a support pedestal, which is located above a leak chase. Since the rack module baseplate is buttressed by the support pedestal and presents a hardened impact surface, this scenario results in a high impact load. The principal design objective is to ensure that the support pedestal does not cause catastrophic damage to the liner and underlying reinforced concrete pool slab, such that rapid loss of pool water occurs. The evaluation shows that the SFP liner will not be pierced, and although the underlying concrete will experience very localized crushing, the SFP structure will not suffer catastrophic damage.

The second scenario considers the load dropping through an empty interior cell near the center of the rack. Since the baseplate is not as stiff at cell locations away from the support pedestal, the principal design objective is to ensure that severing of the baseplate, or large deflection of the baseplate, will not cause the liner to be impacted. The distance from the baseplate to the liner is approximately 5.75 inches. An additional criterion, based on the criticality evaluation, limits the displacement of the dropped assembly and the surrounding eight stored assemblies, to 4 inches. The results of the evaluation show that there is some deformation of the baseplate, as well as localized severing of the baseplate to cell wall welds. The baseplate does not break during the impact. The resulting structural damage has no adverse effect on the coolant flow through the storage

cells. Further, the maximum displacement on the baseplate is 3.36 inches, therefore the liner is not impacted and the criticality acceptance criterion is satisfied. Therefore, the structural consequences are acceptable.

Radiological Considerations

The effect of the proposed increased fuel storage capacity on radiation dose rates in areas adjacent to the SFP and transfer pit was evaluated. Relative to the present racks, the new racks will result in a higher density of fuel next to the SFP walls. The new racks will also be positioned closer to the walls than the present racks. For fuel cooled 72 hours, one year, and five years, the maximum dose rates were calculated to be 12.2, 0.19, and 0.01 mR/hour, respectively.

Although the SFP and transfer pit walls are the same thicknesses, the transfer pit rack would be placed much farther from the transfer pit walls than the SFP racks are placed from the SFP walls. Therefore, the dose rates from the re-racked SFP would bound dose rates resulting from fuel stored in a rack temporarily placed in the transfer pit.

The dose rates at the ceilings of the rooms below the SFP and transfer pit, from the fuel stored in the new racks, will be marginally (probably undetectable) greater than the dose rates from the fuel stored in the present racks. The amount of water, distance, and rack structural metal between active fuel and the floor slabs are greater than between the active fuel and the walls. Therefore, even with the floor slabs being thinner than the walls (5 feet vs. 5.5 feet), the dose rates at the ceilings from the fuel should be no greater than the dose rates through the walls.

It is expected that dose rates experienced in actual practice will be significantly lower than calculated, as the calculated results are based on conservative assumptions. Based on the evaluation, no changes to these radiation zone designations in the USAR are anticipated. During the re-racking, routine radiation surveys will be conducted to determine the actual dose rates in the rooms. Should dose rates above and around the SFP area increase, this change would be identified by routine radiation surveys, and the appropriate radiological controls would be revised as required.

Fuel Handling Area Ventilation System Considerations

As previously discussed, the fuel handling area ventilation system is designed to provide an average of 20 air changes per hour over the surface of the SFP, and to maintain the fuel handling area between 60 and 110 °F. An evaluation of the fuel handling area ventilation was performed for the maximum SFP bulk temperature condition, which is based on the most limiting full core discharge scenario (scenario 4A). The building air temperature in the vicinity of the SFP will be maintained less than or equal to 110 °F, therefore the environmental qualification of essential equipment in the fuel handling building will not be affected.

Technical Specification requirements on the fuel handling area ventilation system ensure that radioactive material released from an irradiated fuel assembly will be filtered through the HEPA and charcoal iodine adsorber filters prior to discharge to the

atmosphere.

SFP Rack Installation Considerations

Section 3.5 of Reference 4 (attached) provides details on heavy load considerations for the proposed rack installation activities. Section 10.0 of Reference 4 provides additional details on installation activities.

The spent fuel cask crane will be used for the installation of the new storage racks in the SFP, and installation and removal of the cask pit impact cover and temporary crane (described below). The spent fuel cask crane is comprised of a main hook rated for 140 tons, as well as an auxiliary hook rated for 20 tons. As described in USAR Section 9.1.5, "Control of Heavy Loads," the spent fuel cask crane, including its auxiliary hoist, is subject to compliance with the applicable guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The maximum load to be lifted during rack installation weighs approximately 17,530 pounds. This load is well within the rating of the spent fuel cask crane hooks. The weight of the cask pit impact cover is also within the rating of the spent fuel cask crane hooks.

As described in Section 3.6 of Reference 4 (attached), due to the limited travel of the spent fuel cask crane, a temporary crane will be used, as necessary, to position existing racks for removal, and for final positioning of the new racks. The crane will be designed to meet the intent of NUREG-0612 through a defense-in-depth approach. The temporary crane will only lift the racks several inches above the pool floor to move them horizontally. It will not be used to lift any heavy loads out of the pool, will not be used to lift any heavy loads over fuel assemblies or safety-related equipment, and will not be used to move fuel assemblies.

The load path of some racks during the re-racking activities may traverse fuel assemblies stored in the cask pit. If it is necessary to move racks over fuel assemblies stored in the cask pit, an impact cover will be required. The physical design of the impact cover, together with administrative controls established while the cover is being moved, ensure that it can not fall into the cask pit in the unlikely event it is dropped. The activities associated with installation and removal of the cover will meet the requirements of NUREG-0612. The cover will be qualified to withstand the drop of the heaviest rack, including rigging. The height that such loads may travel over the cover will be established by calculation based on the design of the cover. Administrative controls will ensure that maximum height and weight restrictions are not exceeded. It is not anticipated that it would be necessary to move racks over fuel assemblies stored in the transfer pit.

The effect of a drop of a spent fuel storage rack was analyzed. The evaluation is described in Section 7.0 of Reference 4 (attached). The heaviest rack was assumed to drop from a height of 46 feet above the SFP or transfer pit floor slabs, impacting the liner plate. The results of the evaluation show that the liner will not be pierced. Although the concrete stratum underneath the liner will sustain localized damage, the impact does not compromise the structural integrity of the SFP or transfer pit. Therefore, an abrupt loss of water will not occur.

Underwater diving operations are required in the SFP to remove underwater obstructions, position the new rack modules, and verify installation per design. Fuel in the SFP will be shuffled, as necessary, to reduce the exposure to the divers. Each diver will be equipped with whole body dosimetry with remote, above surface, readouts that will be continuously monitored by Radiation Protection personnel. Contingency measures will be implemented in the case of signal loss with remote reading dosimeters. Divers will be equipped with extremity dosimetry, and will be equipped with underwater survey instrumentation with remote readout capabilities. Divers will also be in continuous communication with Radiation Protection personnel via a dive master. The DBNPS will conduct radiation surveys of the diving area prior to each diving operation and following the movement of any radioactive components in the SFP. The DBNPS will use either visual or physical barriers to ensure that divers maintain a safe distance from spent fuel assemblies or other high radiation sources stored in the SFP. The DBNPS will also use a safety line attached to the diver and manned by a dive tender at all times.

The DBNPS will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained as low as reasonably achievable (ALARA). Cleanup of source material will be performed, as necessary, in accordance with good ALARA practices. The DBNPS will take appropriate action to maintain water clarity during rack module installation.

Proposed Technical Specification Changes (see attached)

TS 3/4.9.7 Refueling Operations - Crane Travel - Fuel Handling Building, and Associated Bases

The proposed changes to LCO 3.9.7 and SR 4.9.7 provide the same crane travel restriction over fuel assemblies stored in the transfer pit as that currently in place for fuel assemblies stored in the SFP or cask pit. As such, these changes will have no adverse effect on nuclear safety.

The proposed changes to LCO 3.9.7 and SR 4.9.7 also add an asterisked footnote to include an exception to the 2430-pound crane load limitation. This exception would allow an impact cover to be moved over fuel assemblies in the cask pit. This exception would also allow loads in excess of 2430 pounds to be moved over fuel assemblies in the cask pit provided: 1) an impact cover is installed, and 2) administrative controls are established to limit the load to a maximum specified weight of 17,530 pounds and to limit the height that the load may travel over the impact cover. The cover is needed if it is necessary to move racks over fuel assemblies located in the cask pit during re-racking evolutions. The impact cover will be qualified to withstand the drop of the heaviest rack, including rigging, from a maximum permissible height. As such, this exception will have no adverse effect on nuclear safety.

The proposed Bases changes are related to the proposed changes to the associated LCO and SR, and are administrative changes that will have no adverse effect on nuclear safety.

TS 3/4.9.11 Refueling Operations - Storage Pool Water Level

The proposed changes to LCO 3.9.11 and SR 4.9.11 provide the same storage pool water level restriction for fuel assemblies stored in the transfer pit as that currently in place for fuel assemblies stored in the SFP and cask pit. As stated in the associated Bases, the water level restriction ensures that sufficient water depth is available to remove activity released from the rupture of an irradiated fuel assembly. As such, these changes will have no adverse effect on nuclear safety.

TS Bases 3/4.9.10 and 3/4.9.11 Water Level - Reactor Vessel and Storage Pool

The proposed Bases change is an administrative change, which clarifies the intended meaning of the discussion regarding iodine, gap activity. This administrative change will have no adverse effect on nuclear safety.

TS 3/4.9.12 Refueling Operations - Storage Pool Ventilation

The proposed changes to LCO 3.9.12 extend the same requirements regarding storage pool ventilation to irradiated fuel located within the transfer pit. As stated in the associated Bases, the storage pool ventilation requirements ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. As such, these changes will have no adverse effect on nuclear safety.

TS 3/4.9.13 Refueling Operations - Spent Fuel Assembly Storage, and Associated Bases

The proposed changes to LCO 3.9.13 and SR 4.9.13.1 apply the appropriate restrictions to fuel assemblies stored in the transfer pit. As stated in the associated Bases, the restrictions regarding fuel assembly storage are consistent with the criticality safety analyses performed for the spent fuel storage racks. As such, these changes will have no adverse effect on nuclear safety.

The proposed changes to the titles of Figures 3.9-1 and 3.9-2 are administrative clarifications and will have no adverse effect on nuclear safety.

The proposed new Figure 3.9-3, "Burnup vs. Enrichment Curves For the Davis-Besse High Density Spent Fuel Pool and Transfer Pit Storage Racks," incorporates new requirements specific to the new spent fuel storage racks. The figure provides burnup/enrichment limitations appropriate for the spent fuel storage rack configurations, consistent with the criticality safety analyses performed. As such, this proposed change will have no adverse effect on nuclear safety.

The same rigorous controls presently applied to fuel movements in the spent fuel pool and cask pit will also be applied to fuel movements in the transfer pit, to ensure that the basis for TS 3.9.13 will be preserved. These controls include:

- Preparation and independent review of all fuel movement sheets for compliance with TS 3.9.13 by the Nuclear Engineering Unit.

- Reactor Engineering oversight of Operations during all fuel movements.

- Independent verification of refueling device (bridge, crane, etc.) location prior to fuel assembly placement or retrieval in the spent fuel storage racks.

- Visual verification that the spent fuel storage rack loading pattern for those assemblies moved complies with TS 3.9.13 within 30 days of any fuel movement in the spent fuel storage racks.

- Chemistry verification every 72 hours that the SFP/cask pit/transfer pit boron concentration is at least 1800 ppm during fuel movements in the SFP, cask pit, and transfer pit, and until the spent fuel storage rack loading pattern verification is performed.

Bases changes are proposed that relate to the proposed changes to the associated LCO. An additional Bases change is proposed that clarifies that the term "directly adjacent" as used in Figure 3.9-1 refers to fuel assemblies stored face-to-face. These Bases changes are administrative changes that will have no adverse effect on nuclear safety.

<u>TS 5.6 Design Features - Fuel Storage</u>

The proposed revision of TS 5.6.1.1, clarifying that this section applies to the current low density spent fuel racks located in the SFP, is an administrative change and will have no adverse effect on nuclear safety.

The proposed revision of TS 5.6.1.3, clarifying that this section applies to the current high density spent fuel racks located in the cask pit, is an administrative change that will have no adverse effect on nuclear safety. The revisions to this same section, expanding the scope to include the high density spent fuel racks located in the SFP and the transfer pit, describes the criticality design features for the same new spent fuel storage racks, and places restrictions for the design and maintenance of these racks in the TS. These restrictions ensure that the evaluations in Reference 4 remain valid and, thus, will have no adverse effect on nuclear safety.

The proposed change to TS 5.6.2 to include the transfer pit as a fuel storage area, creates a new requirement to ensure the transfer pit, similar to the spent fuel storage pool and cask pit, does not inadvertently drain below 9 feet above the top of the fuel storage racks. This requirement ensures that the evaluations in Reference 4 remain valid and, thus, will have no adverse effect on nuclear safety.

The proposed changes to TS 5.6.3 include a description of the storage capacity of the transfer pit, as well as a description on the total number of fuel assemblies that may be stored in the SFP, cask pit, and transfer pit collectively. These changes ensure that the evaluations in Reference 4 remain valid and, thus, will have no adverse effect on

nuclear safety.

Conclusion

Based on the technical basis described in Reference 4 (attached), as summarized above, and based on the above evaluation of each individually proposed TS change, it is concluded that the proposed changes, including the expanded SFP storage capacity resulting from the planned re-racking of the SFP, and the inclusion of provisions allowing for temporary storage of fuel assemblies in the transfer pit, will have no adverse effect on nuclear safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because the methods and procedures for handling fuel assemblies will remain unchanged, fuel handling equipment reliability will be unaffected, and provisions will remain in place to ensure that the likelihood of a heavy load drop will remain extremely small. The proposed changes involve an expanded SFP storage capacity resulting from the planned re-racking of the SFP, and the inclusion of provisions allowing for temporary storage of fuel assemblies in the transfer pit.

For the installation activities involving the proposed expanded spent fuel storage capacity, heavy load lifts have been given careful consideration. In accordance with the proposed changes to Technical Specification (TS) 3/4.9.7, "Crane Travel - Fuel Handling Building," except when a specially designed impact cover is placed over fuel assemblies located in the cask pit, heavy loads are prohibited from travel over stored fuel assemblies. The physical design of the impact cover, together with administrative controls established while the impact cover is being installed or removed, ensure that it can not fall into the cask pit in the unlikely event that it is dropped. As described below, except for the use of a temporary crane, the spent fuel cask crane will be used for the replacement of the existing storage racks in the spent fuel pool (SFP), placement of the temporary rack in the transfer pit, and eventual relocation of racks from the cask pit and transfer pit to the SFP. The spent fuel cask crane is comprised of a main hook rated for 140 tons, as well as an auxiliary hook rated for 20 tons. As described in the DBNPS Updated Safety Analysis Report (USAR) Section 9.1.5, "Control of Heavy Loads," the spent fuel cask crane, including its auxiliary hoist, is subject to compliance

with the applicable guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This will ensure that there will be no significant increase in the probability of a heavy load drop, and that the probability of a heavy load drop will remain extremely small. Due to the limited travel of the spent fuel cask crane, a temporary crane will be used, as necessary, to position existing racks for removal and for final positioning of the new racks. The crane will be designed to meet the intent of NUREG-0612 through a defense-in-depth approach. The temporary crane will only lift the racks several inches above the pool floor to move them horizontally. It will not be used to lift any heavy load out of the pool, will not be used to lift any heavy loads over fuel assemblies or safety-related equipment, and will not be used to move fuel assemblies. The methods and procedures for handling fuel assemblies during installation activities will not be significantly changed. Based on these considerations, there will be no significant increase in the probability of damage to stored fuel assemblies as a result of installation activities.

For the activities involving the post-installation use of the proposed expanded spent fuel storage capacity, the following previously postulated accident scenarios have been considered: Misloaded or Mislocated Fuel Assembly; Seismic Event; and Fuel Handling Accident. In addition, the effects of a loss of spent fuel pool cooling or level have been evaluated. The probability of the inadvertent misloading or mislocation of a fuel assembly is primarily a function of fuel handling procedures. The probability of a fuel handling accident is primarily a function of fuel handling equipment reliability and fuel handling procedures. The methods and procedures for handling fuel assemblies during normal, post-installation use of the racks will not be significantly changed. In addition, following completion of installation activities, the activities performed in and around the spent fuel pool will not be significantly changed due to the use of the new spent fuel pool racks. The proposed TS changes have no bearing on the probability of a seismic event or the probability of a loss of spent fuel pool cooling or level. Based on these considerations, there will be no significant increase in the probability of an accident previously evaluated as a result of normal, post-installation use of the racks.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because evaluations for each postulated accident have shown that the consequences remain bounded by the consequences from the previously evaluated accidents.

For the installation activities involving the proposed expanded spent fuel storage capacity, heavy load lifts have been given careful consideration. Heavy load lifts are subject to compliance with the applicable guidelines of NUREG-0612. These guidelines include use of defined safe load paths in accordance with approved procedures. This will ensure that there will be no significant increase in the consequences of a heavy load drop, in the unlikely event that one were to occur.

For the activities involving the post-installation use of the proposed expanded spent fuel storage capacity, the following previously postulated accident scenarios have been considered: Misloaded or Mislocated Fuel Assembly; Seismic Event; and Fuel Handling Accident. In addition, the effects of a loss of spent fuel pool cooling or level have been evaluated. The criticality analyses for the new spent fuel pool storage racks

require burnup/enrichment limitations similar to those currently in place for the existing racks. These burnup/enrichment limitations are imposed by the proposed changes to TS 3/4.9.13, Refueling Operations – Spent Fuel Assembly Storage. The criticality evaluation for the new racks shows that if an unirradiated fuel assembly of the highest permissible enrichment is placed in an unauthorized storage cell or mislocated outside a storage rack, k_{eff} will be maintained ≤ 0.95, taking credit for soluble boron in the spent fuel pool water. Therefore, there will be no adverse radiological consequences due to the proposed changes.

The results of the seismic evaluation demonstrate that the racks will remain intact and that the structural capability of the pool and liner will not be exceeded. The Auxiliary Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks and pool water inventory, therefore, the rack geometry and cooling to the fuel will be maintained. Thus, there will be no adverse radiological consequences due to the proposed changes.

The new racks do not change the height of the stored fuel relative to any load being handled, and the 72 hour decay time for the fuel assumed in the design basis accident is conservative. Based on this, the design basis fuel handling accident for the pool area remains unchanged.

The mechanical accidents analyses evaluated the extent of rack deformation due to different scenarios. Based on the maximum calculated rack deformation, it was concluded that the criticality and thermal hydraulic limitations were not exceeded. Also, the mechanical accidents analyses concluded that the pool liner will not be pierced, and there will be no catastrophic damage to the pool structure. Therefore, the analyzed mechanical accidents will not lead to radiological consequences beyond that already evaluated.

The evaluation of a loss of spent fuel pool cooling shows that sufficient time will be available, before a significant reduction in water level, to restore cooling or to provide a source of makeup water. Therefore, the racks will remain submerged and fuel stored therein will remain sufficiently cooled, and there will be no adverse radiological consequences due to the proposed changes.

The fuel handling area ventilation system will continue to ensure that in the event radioactive material is released from a damaged irradiated fuel assembly, it will be filtered through HEPA and charcoal iodine adsorber filters prior to discharge to the atmosphere. Therefore, the radiological consequences will continue to be mitigated as prior to the proposed changes.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the function and parameters of the components and the associated activities necessary to support safe storage of fuel assemblies in the new racks are similar to those presently in place. The methods and procedures for handling fuel assemblies would not be changed. Therefore, the list of postulated accidents remains unchanged.

Any event which would modify parameters important to safe fuel storage sufficiently to place them outside of the boundaries analyzed for normal conditions and/or outside of the boundaries previously considered for accidents would be considered a new or different accident. The fuel storage configuration and the existence of the coolant are the parameters that are important to safe fuel storage. The proposed changes do not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents, nor do they affect the important parameters required to ensure safe fuel storage. Therefore, the potential for a new or previously unanalyzed accident is not created.

3. Not involve a significant reduction in a margin of safety because for the proposed changes, appropriate evaluations have shown compliance with stipulated safety margins.

   The objective of spent fuel storage is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as a seismic event or a fuel handling accident. The design of the new spent fuel racks meets all applicable requirements for safe fuel storage. The seismic and structural design of the racks preserves the proper margin of safety during normal and abnormal loads. The methodology used in the criticality analysis meets the applicable regulatory guidance. The thermal-hydraulic evaluation demonstrates that the pool will be maintained below the specified thermal limits under the conditions of the maximum heat load and during all credible malfunction scenarios and seismic events. Upon the unlikely event of a complete loss of spent fuel pool cooling, sufficient time will be available, before a significant reduction in water level, to restore cooling or to provide a source of makeup water. Therefore, the racks will remain submerged and fuel stored therein will remain sufficiently cooled. In addition, the results of the fuel handling accident evaluation show that the minimum subcriticality margin will be maintained, cooling will remain adequate, the spent fuel pool structure will not suffer catastrophic damage, and the radiological dose resulting from the release caused by a fuel handling accident will not be increased from that previously considered.

   Thus, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

CONCLUSION:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENT:

Attached are the proposed marked-up changes to the Operating License.

REFERENCES:

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 243.

2. DBNPS Updated Safety Analysis Report through Revision 21.

3. 10 CFR 50.59, "Changes, Tests, and Experiments."

4. "Design and Licensing Report, Davis-Besse Spent Fuel Pool Rerack Project," Holtec International Report No. HI-992329, Revision 1 (see Attachments 4 and 5).

5. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 167 to Facility Operating License No. DPR-77 and Amendment No. 157 to Facility Operating License No. DPR-79, Tennessee Valley Authority, Sequoyah Nuclear Plant, Units 1 and 2, Docket Nos. 50-327 and 50-328," dated April 28, 1993 (TAC Nos. M83068 and M83069).

6. American Concrete Institute (ACI) 349-85 and 349R-85, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary."

7. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July, 1980.

8. DBNPS letter dated May 21, 1999 (DBNPS Serial Number 2550), "License Amendment Application to Allow Use of Expanded Spent Fuel Storage Capability (License Amendment Request No. 98-0007; TAC No. MA5477)," as supplemented by letters dated December 1, 1999 (DBNPS Serial Number 2628) and January 28, 2000 (DBNPS Serial Number 2641).

9. NRC License Amendment Number 237 (TAC No. MA5477) to DBNPS Facility Operating License No. NPF-3, dated February 29, 2000 (DBNPS Log Number 5614).

10. DBNPS letter dated July 28, 1999 (DBNPS Serial Number 2397), "License Amendment Application to Revise Technical Specification 3/4.7.5.1, Ultimate Heat Sink (License Amendment Request No. 96-0008; TAC No. MA6092)," as supplemented by letter dated June 6, 2000 (DBNPS Serial Number 2654).

11. NRC License Amendment Number 242 (TAC No. MA6092) to DBNPS Facility Operating License No. NPF-3, dated September 12, 2000 (DBNPS Log Number 5705).

12. Holtec Report HI-981864 Revision 5, DB Vendor Calculation No. C-63Q-1.

13. NRC License Amendment Number 129 (TAC No. MA1113) to Callaway Plant, Unit 1, Operating License No. NPF-30, dated February 4, 1999.

REFUELING OPERATIONS

CRANE TRAVEL - FUEL HANDLING BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2430 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool, ~~or in the~~ cask pit*, or transfer pit.

APPLICABILITY: With fuel assemblies and water in the spent fuel pool, ~~or in the~~ cask pit, or transfer pit.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The weight of each load, other than a fuel assembly, shall be verified to be ≤ 2430 pounds prior to moving it over fuel assemblies in the spent fuel pool, ~~or~~ cask pit*, or transfer pit.

* An impact cover weighing in excess of 2430 pounds may be moved over fuel assemblies in the cask pit provided that administrative controls are established. Other loads in excess of 2430 pounds may be moved over fuel assemblies in the cask pit provided: 1) an impact cover is installed, and 2) administrative controls are established to limit the load to 17,530 pounds and to limit the height that the load may travel over the impact cover.

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks in the spent fuel pool, ~~or~~ cask pit, or transfer pit.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool, ~~or~~ cask pit, or transfer pit.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel and crane operations with loads in the fuel storage area and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pool, ~~and~~ cask pit, and transfer pit shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in these locations.

REFUELING OPERATIONS

STORAGE POOL VENTILATION

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent emergency ventilation systems servicing the storage pool area shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool, ~~or~~ cask pit, or transfer pit.

ACTION:

a. With one emergency ventilation system servicing the storage pool area inoperable, fuel movement within the spent fuel pool, ~~or~~ cask pit, or transfer pit, or crane operation with loads over the spent fuel pool, ~~or~~ cask pit, or transfer pit, may proceed provided the OPERABLE emergency ventilation system servicing the storage pool area is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no emergency ventilation system servicing the storage pool area OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool, ~~or~~ cask pit, or transfer pit, or crane operation with loads over the spent fuel pool, ~~or~~ cask pit, or transfer pit, until at least one system is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12.1 The above required emergency ventilation system servicing the storage pool area shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.6.5.1, and at least once each REFUELING INTERVAL by verifying that the emergency ventilation system servicing the storage pool area maintains the storage pool area at a negative pressure of $\geq 1/8$ inches Water Gauge relative to the outside atmosphere during system operation.

4.9.12.2 The normal storage pool ventilation system shall be demonstrated OPERABLE at least once each REFUELING INTERVAL by verifying that the system fans stop automatically and that dampers automatically divert flow into the emergency ventilation system on a fuel storage area high radiation test signal.

REFUELING OPERATIONS

SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 Fuel assemblies shall be placed in the spent fuel storage racks in accordance with the following criteria:

a. Fuel assemblies stored in the spent fuel pool shall ~~be placed in the spent fuel storage racks in accordance with~~ meet the criteria shown in Figure 3.9-1, when located in the low density spent fuel storage racks.

b. Fuel assemblies stored in the cask pit shall ~~be placed in the spent fuel storage racks in accordance with~~ meet the criteria shown in Figure 3.9-2, when located in the high density spent fuel storage racks.

c. Fuel assemblies stored in the spent fuel pool or transfer pit shall meet the criteria shown in Figure 3.9-3, when located in the high density spent fuel storage racks.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool, ~~or~~ cask pit, or transfer pit.

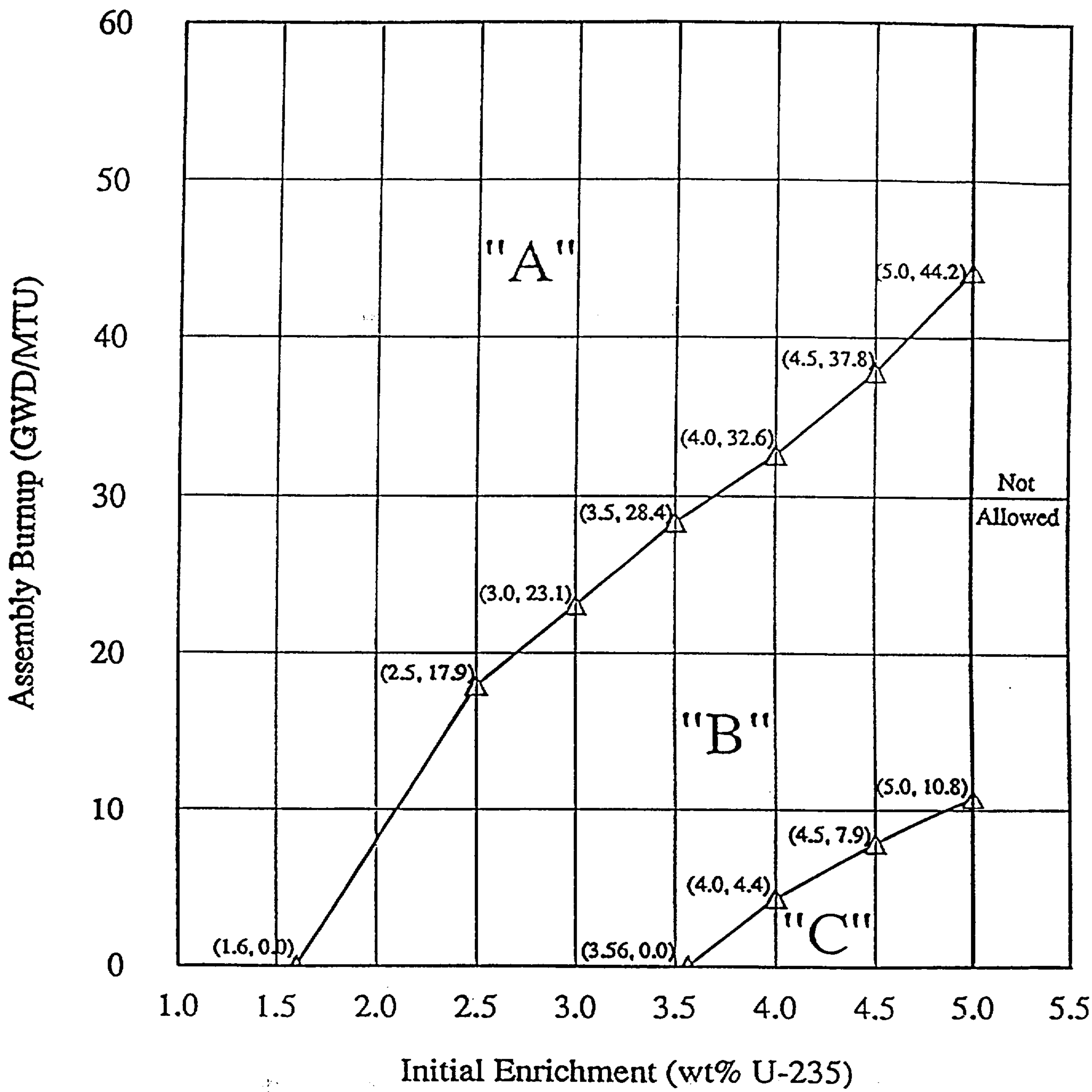
ACTION:

With the requirements of the above ~~Specifications~~ specification ~~3.9.13.a or 3.9.13.b~~ not satisfied, suspend all other fuel movement within the spent fuel pool, ~~or~~ cask pit, or transfer pit and move the non-complying fuel assemblies to allowable locations in accordance with Figure 3.9-1 ~~for the spent fuel pool, or~~, Figure 3.9-2, or Figure 3.9-3 ~~for the cask pit~~, as appropriate. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

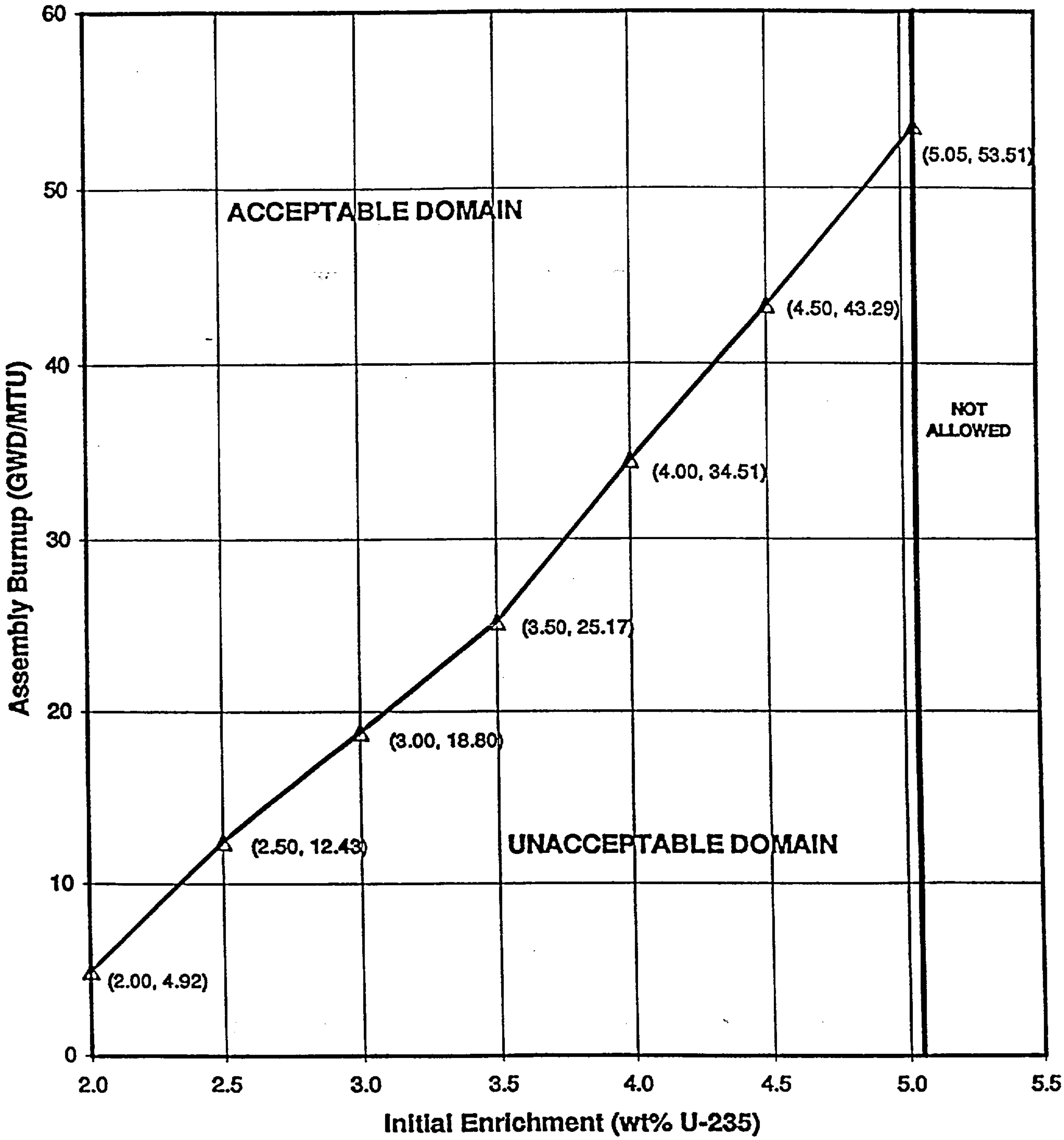
4.9.13.1 Prior to storing a fuel assembly in the spent fuel pool, ~~or~~ cask pit, or transfer pit, verify by administrative means that the initial enrichment and burnup of the fuel assembly are in accordance with Figure 3.9-1 ~~for the spent fuel pool~~, ~~or~~ Figure 3.9-2, or Figure 3.9-3 ~~for the cask pit~~, as appropriate.

Figure 3.9-1
Burnup vs. Enrichment Curves
For the Davis-Besse Low Density
Spent Fuel Pool Storage Racks



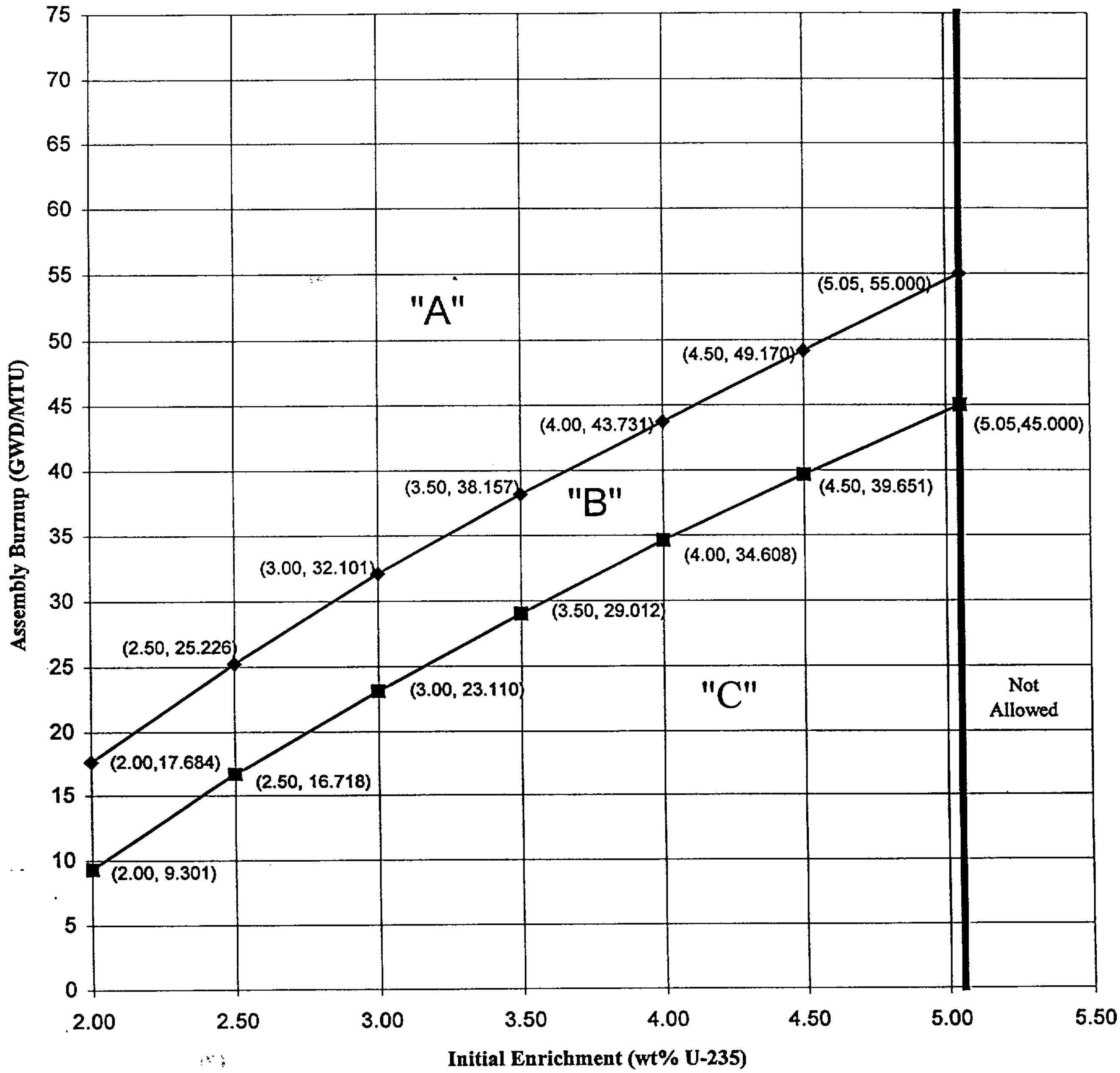
Category "A": May be placed in any rack location
Category "B": Must not be placed directly adjacent to Category "C" assemblies
Category "C": May only be placed directly adjacent to Category "A" assemblies or non-fuel locations

Figure 3.9-2
Burnup vs. Enrichment Curve
For the Davis-Besse High Density
Cask Pit Storage Racks



Note: Fuel assemblies with initial enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies).

Figure 3.9-3
Burnup vs. Enrichment Curves
For the Davis-Besse High Density
Spent Fuel Pool and Transfer Pit Storage Racks



Notes: Fuel assemblies with initial enrichments less than 2 wt% U-235 will conservatively be required to meet the burnup requirements of 2.0 wt% U-235 assemblies.
Loading pattern considerations applicable to Category "A", "B", and "C" assemblies are described in the Bases.

Revised by NRC letter dated:
August 12, 1994

3/4.9.6 FUEL HANDLING BRIDGE OPERABILITY

The OPERABILITY requirements of the hoist bridges used for movement of fuel assemblies ensures that: 1) fuel handling bridges will be used for movement of control rods and fuel assemblies, 2) each hoist has sufficient load capacity to lift a fuel element, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly in a failed fuel container over other fuel assemblies in the spent fuel pool, ~~or~~ cask pit, or transfer pit ensures that in the event this load is dropped (1) the activity release will not exceed the source term assumed in the design basis fuel handling accident for outside containment, and (2) any possible distortion of fuel in the storage racks will not result in a critical array.

During spent fuel pool re-racking activities, if it is necessary to move a storage rack over fuel assemblies stored in the cask pit, the 2430 pound weight limitation may be exceeded in order to install or remove an impact cover over the cask pit. The physical design of the impact cover, together with administrative controls established while the cover is being moved, ensure that it can not fall into the cask pit in the unlikely event that it is dropped. Once installed over the cask pit, the impact cover is capable of withstanding a dropped load of up to 17,530 pounds (the heaviest rack, including rigging). The height that such loads may travel over the cover is established by calculation based on the design of the cover. Administrative controls ensure that maximum height and weight restrictions are not exceeded.

3/4.9.8 COOLANT CIRCULATION

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two DHR loops OPERABLE when there is less than 23 feet of water above the core ensures that a single failure of the operating DHR loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating DHR loop, adequate time is provided to initiate emergency procedures to cool the core.

In MODE 6, the RCS boron concentration is typically somewhat higher than the boron concentration required by Specification 3.9.1, and could be higher than the boron concentration of normal sources of water addition. The flowrate through the decay heat system may at times be reduced to somewhat less than 2800 gpm. In this situation, if water with a boron concentration equal to or greater than the boron concentration required by Specification 3.9.1 is added to the RCS, the RCS is assured to remain above the Specification 3.9.1 requirement, and a flowrate of less than 2800 gpm is not of concern.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

Deleted

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the ~~assumed 10%~~ iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 STORAGE POOL VENTILATION

The requirements on the emergency ventilation system servicing the storage pool area to be operating or OPERABLE ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

The restrictions on the placement of fuel assemblies within the spent fuel pool, ~~and~~ cask pit, and transfer pit, as dictated by Figure 3.9-1, ~~and~~ Figure 3.9-2, and Figure 3.9-3, ensure that the k-effective of the spent fuel pool, ~~and~~ cask pit, and transfer pit will always remain less than 0.95 assuming the spent fuel pool, ~~and~~ cask pit, and transfer pit to be flooded with non-borated water. The restrictions delineated in Figure 3.9-1, ~~and~~ Figure 3.9-2, and Figure 3.9-3, and the action statement, are consistent with the criticality safety analyses performed for the spent fuel pool, ~~and~~ cask pit, and transfer pit. The term "directly adjacent" as used in Figure 3.9-1 refers to fuel assemblies stored face-to-face.

The criticality analyses qualify the high density rack modules for storage of fuel assemblies in one of three different loading patterns, subject to certain restrictions: Mixed Zone Three Region, Checkerboard, and Homogeneous Loading. Figure 3.9-3 provides the Category-specific burnup/enrichment limitations. Different loading patterns may be used in different rack modules, provided each rack module contains only one loading pattern. Two different loading patterns may be used in a single rack module, subject to certain additional restrictions. The loading pattern restrictions are maintained in fuel handling administrative procedures.

The design features of the low density spent fuel storage racks are described in Specification 5.6.1.1. The design features of the high density spent fuel storage racks are described in Specification 5.6.1.3.

5.0 DESIGN FEATURES

5.1 Site Location

The Davis-Besse Nuclear Power Station, Unit Number 1, site is located on Lake Erie in Ottawa County, Ohio, approximately six miles northeast from Oak Harbor, Ohio and 21 miles east from Toledo, Ohio. The exclusion area boundary has a minimum radius of 2400 feet from the center of the plant.

5.2 (Deleted)

5.3 Reactor Core

5.3.1 Fuel Assemblies

The reactor core shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zircaloy, M5, or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

5.3.2 Control Rods

The reactor core shall contain 53 safety and regulating control rod assemblies and 8 axial power shaping rod (APSR) assemblies. The nominal values of absorber material for the safety and regulating control rods shall be 80 percent silver, 15 percent indium and 5 percent cadmium. The absorber material for the APSRs shall be 100 percent Inconel.

5.4 (Deleted)

5.5 (Deleted)

5.6 Fuel Storage

5.6.1 Criticality

5.6.1.1 The low density spent fuel pool storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1% delta k/k for calculation uncertainty.

(continued)

5.6 Fuel Storage (continued)

b. A rectangular array of stainless steel cells spaced 12 31/32 inches on centers in one direction and 13 3/16 inches on centers in the other direction. Fuel assemblies stored in the spent fuel pool shall be placed in a stainless steel cell of 0.125 inches nominal thickness or in a failed fuel container.

c. Fuel assemblies stored in the spent fuel pool in accordance with Technical Specification 3.9.13.

5.6.1.2 The new fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1% delta k/k for uncertainties as described in Section 9.1 of the USAR.

b. A K_{eff} equivalent to less than or equal to 0.98 when immersed in a hydrogenous "mist" of such a density that provides optimum moderation (i.e., highest value of K_{eff}), which includes a conservative allowance of 1% delta k/k for uncertainties as described in Section 9.1 of the USAR.

c. A nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks.

d. Fuel assemblies having a maximum initial enrichment of 5.0 weight percent uranium-235.

5.6.1.3 The <u>high density spent fuel pool storage racks,</u> cask pit storage racks<u>, and transfer pit rack</u> are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for manufacturing tolerances and calculation uncertainty.

b. A rectangular array of stainless steel cells <u>with walls of 0.075 inches nominal thickness,</u> spaced a nominal 9.22 inches on center in both directions. Boral neutron absorber material is utilized between each cell for criticality considerations. ~~Fuel assemblies stored in the cask pit shall be placed in a stainless steel cell with walls of 0.075 inches nominal thickness.~~

c. Fuel assemblies stored in the <u>spent fuel pool,</u> cask pit<u>, or transfer pit</u> in accordance with Technical Specification 3.9.13.

5.6 Fuel Storage (continued)

5.6.2 Drainage

The spent fuel storage pool, ~~and~~ cask pit, and transfer pit are designed and shall be maintained to prevent inadvertent draining below 9 feet above the top of the fuel storage racks.

5.6.3 Capacity

a. The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~735~~ 1624 fuel assemblies, less the number of fuel assemblies stored in racks located in the cask pit and transfer pit.

b. The cask pit is designed and shall be maintained with a storage capacity limited to no more than 289 fuel assemblies.

c. The transfer pit is designed and shall be maintained with a storage capacity limited to no more than 90 fuel assemblies.

5.7 (Deleted)

**ENVIRONMENTAL ASSESSMENT
FOR
LICENSE AMENDMENT REQUEST NUMBER 98-0013**

Identification of Proposed Action

This proposed action involves the Davis-Besse Nuclear Power Station (DBNPS), Unit 1, Operating License Number NPF-3, Appendix A, Technical Specifications (TS). The proposed license amendment application involves TS 3/4.9.7, Refueling Operations - Crane Travel - Fuel Handling Building, and associated Bases; TS 3/4.9.11, Refueling Operations - Storage Pool Water Level, and associated Bases; TS 3/4.9.12, Refueling Operations - Storage Pool Ventilation; TS 3/4.9.13, Refueling Operations - Spent Fuel Assembly Storage, and associated Bases; and TS 5.6, Design Features - Fuel Storage.

The proposed TS changes would expand the present spent fuel pool storage capability from the current capacity of 735 fuel assemblies, to a capacity of 1624 fuel assemblies by replacing all of the existing racks with high density racks. To provide additional temporary storage of fuel assemblies to support a complete re-racking of the SFP, the license amendment application also requests approval for up to 90 transfer pit storage locations. The transfer pit storage rack will be relocated into the SFP as part of the completion of this re-racking project.

Need for the Proposed Action

A facility for the long-term storage of spent nuclear fuel assemblies from commercial nuclear power reactors is to be provided by the United States Department of Energy. However, since such a facility is not yet available or expected to be available until at least the year 2010, commercial nuclear power plants, such as the DBNPS, have had to provide for additional spent fuel storage.

The DBNPS began operating Cycle 12 (May, 1998) with insufficient storage capacity in the spent fuel pool (SFP) to fully offload the entire reactor core (177 fuel assemblies). Since a full core offload into the SFP was required for the performance of the ten-year Inservice Inspection activities during the Spring, 2000 Twelfth Refueling Outage (12RFO), the DBNPS submitted License Amendment Request (LAR) 98-0007 (DBNPS Serial Number 2550) on May 21, 1999, to allow the use of spent fuel racks in the cask pit area adjacent to the SFP. License Amendment Number 237 was issued on February 29, 2000, providing approval for use of up to 289 cask pit rack storage locations. As described in LAR 98-0007, this added storage capability will also be utilized to provide temporary storage of fuel to support a complete re-racking of the SFP, and the four cask pit storage racks will be relocated into the SFP as part of the final completion of this re-racking project.

The new SFP fuel storage capacity of 1624 fuel assemblies will be sufficient to meet storage needs through the current expiration date of the DBNPS operating license, April 22, 2017.

Environmental Impact of the Proposed Action

As described in the Safety Assessment and Significant Hazards Consideration (SASHC) for the proposed license amendment application, the DBNPS has determined that the structures, systems, and components which could be affected by the proposed license amendment, will continue to be capable of performing their safety functions.

The proposed license amendment application involves a change to a requirement with respect to the use of plant components located within the restricted area as defined in 10 CFR Part 20. As concluded in the SASHC, this proposed license amendment does not involve a significant hazards consideration. The proposed changes to expand the fuel storage capability in the SFP, and allow temporary storage of fuel assemblies in the transfer pit, do not alter source terms, containment isolation, or allowable releases. In addition, as described in further detail below, the proposed changes do not involve a significant change in the types or a significant increase in the amounts of any radiological effluents that may be allowed to be released offsite. Furthermore, as also described in further detail below, there is no significant increase in the individual or cumulative occupational radiation exposure.

The spent fuel pool cooling and cleanup system currently generates approximately 50 cubic feet of solid radioactive waste annually. The necessity for pool filtration resin replacement is determined primarily by the need for water clarity, and the resin is normally changed about once every 18 months. Re-racking activities may result in a one-time shortening of the resin change-out interval or an increase in filter usage, however, the long-term normal resin and filter replacement frequency is not expected to be significantly affected by the additional number of fuel assemblies in storage.

Although no significant increase in the annual volume of solid radioactive waste is expected from operating with expanded spent fuel storage capacity, there are 12 fuel storage rack modules and a module for 15 failed fuel storage locations currently installed in the SFP that are being replaced with the new rack modules. There are also other miscellaneous items in the SFP, such as portions of piping, which will be removed to accommodate the new racks. There will be a one-time increase in solid waste generation due to the need to dispose of these components, however this represents an insignificant incremental increase in the total quantity of solid waste generated as a result of plant operation. The old racks and other miscellaneous items will be decontaminated underwater via pressure washing or other acceptable cleaning mechanisms, prior to removal from the pool area. The rack modules will be disassembled as required to facilitate their removal from the pool. The components will be removed from the pool under Radiation Protection dose rate surveillance, and transported to a designated location for any needed wrapping or placement into anti-contamination bags. An appropriate shipping container will be used to remove the existing rack components for eventual processing.

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

The contribution of gaseous releases from the fuel storage area is negligible in comparison to other releases, and no significant increase due to the increased quantity of stored spent fuel assemblies is expected. The discharge of gaseous radioactive effluents will continue to be a small fraction of regulatory limits.

During normal operations, personnel working in the fuel storage area are exposed to radiation from the SFP. Operating experience has shown that area radiation dose rates originate primarily from radionuclides in the pool water. During refueling and other fuel movement operations, pool water concentrations might be expected to increase somewhat due to crud deposits spalling from fuel assemblies and due to activities carried into the pool from the primary system. Fuel movement operations as a result of rack installation activities may marginally increase dose rates above and around the SFP and cask pit perimeter. However, the dose fields should still approximate conditions seen during normal operating conditions. Should dose rates above and around the SFP area increase, this change would be identified by routine radiation surveys, and the appropriate radiological controls would be revised as required.

Radiation dose rates in accessible areas around the spent fuel storage and transfer zones have been conservatively evaluated based on realistic fuel parameters. Dose rates will remain within regulatory limits. No changes to the radiation zone designations described in the DBNPS Updated Safety Analysis Report (USAR) are anticipated.

Operating experience has also shown that there have been negligible concentrations of airborne radioactivity in the SFP area. No increase in airborne radioactivity is expected as a result of the expanded storage capacity.

The proposed change does not result in new surveillances which would require additional personnel entry into radiation controlled areas.

The existing radiation protection program at the DBNPS is adequate for the rack removal/installation operations. Personnel doses, including diving operations, will be maintained as low as reasonably achievable (ALARA). Personnel exposure is estimated to be no greater than 12 person-rem.

With regard to potential non-radiological impacts, the proposed license amendment involves no significant increase in the amounts or change in the types of any non-radiological effluents that may be released offsite, and has no other environmental impact.

Based on the above, the DBNPS concludes that there are no significant radiological or non-radiological environmental impacts associated with the proposed license amendment.

Alternatives to the Proposed Action

Since the DBNPS has concluded that the environmental effects of the proposed action are not significant, any alternatives will have only similar or greater environmental impacts. The principal alternative would be to not grant the license amendment. Since the environmental impacts of the proposed action are not significant, denial of the proposed license amendment would not significantly reduce the environmental impacts attributable to the plant.

Alternative Use of Resources

This action does not involve the use of resources not previously considered in the Final Environmental Statement Related to the Operation of the Davis-Besse Nuclear Power Station, Unit Number 1 (NUREG 75/097).

Finding of No Significant Impact

The DBNPS has reviewed the proposed license amendment against the categorical exclusion criteria of 10 CFR 51.22(c)(9) for an environmental assessment. As demonstrated in the proposed license amendment's SASHC, the proposed changes do not involve a significant hazards consideration. In addition, the proposed changes do not significantly change the types or significantly increase the amounts of effluents that may be released offsite, and do not significantly increase individual or cumulative occupational radiation exposures. Accordingly, the DBNPS finds that the proposed license amendment, if approved by the Nuclear Regulatory Commission, will have no significant impact on the environment and that no environmental assessment is required.

Docket Number 50-346
License Number NPF-3
Serial Number 2640
Attachment 3

AFFIDAVIT PURSUANT TO 10 CFR 2.790

(4 pages follow)

HOLTEC
INTERNATIONAL
Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Scott H. Pellet, being duly sworn, depose and state as follows:

(1) I am the Project Manager for Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the document entitled "Design and Licensing Report, Davis-Besse Spent Fuel Reracking Project," Holtec Report HI-992329, revision 1. The proprietary material in this document is delineated by proprietary designation (i.e., shaded text) on pages 4-9, 4-26, 5-9, 5-10, 5-22, 5-23, 6-24, 6-25, and 6-27.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

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Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
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Fax (856) 797-0909

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(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the

HOLTEC
INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
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Fax (856) 797-0909

AFFIDAVIT PURSUANT TO 10CFR2.790

correct analytical methodology is difficult to quantify, but it clearly is substantial. Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Scott H. Pellet, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 21st day of November, 2000.

Scott H. Pellet

Mr. Scott H. Pellet
Holtec International

Subscribed and sworn before me this 21st day of November ~~, 2000~~.

Maria C. Pepe

NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005