February 29, 2000 Template # NRR-058

Mr. Guy G. Campbell, Vice President - Nuclear
FirstEnergy Nuclear Operating Company
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: ISSUANCE OF AMENDMENT - DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 (TAC NO. MA5477)

Dear Mr. Campbell:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No237 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit 1. The amendment is in response to your application dated May 21, 1999 (License Amendment Request No. 98-0007, Serial Number 2550) and supplemented by submittals dated December 1, 1999 (Serial Number 2628) and January 28, 2000 (Serial Number 2641).

This amendment revises the Technical Specifications (TSs) to expand the present spent fuel storage capability by 289 storage locations by allowing the use of spent fuel racks in the cask pit area adjacent to the spent fuel pool. The changes include 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"; TS 3/4.9.12, "Refueling Operations - Storage Pool Ventilation"; TS 3/4.9.13, "Refueling Operations - Spent Fuel Pool Fuel Assembly Storage;" and associated Bases; TS 5.6, "Design Features - Fuel Storage"; and related changes to the TS Index.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No.237to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. NPF-3

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee) dated May 21, 1999, and supplemented by submittals dated December 1, 1999, and January 28, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.237, are hereby incorporated in the license. FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 120 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 29, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 237

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
VIII	VIII
XIII	XIII
XIV	XIV
3/4 9-7	3/4 9-7
3/4 9-11	3/4 9-11
3/4 9-12	3/4 9-12
3/4 9-13	3/4 9-13
3/4 9-14	3/4 9-14
- - -	3/4 9-15
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
5-1	5-1
5-2	5-2
- - -	5-3

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

SECTION

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.

APPLICABILITY: With fuel assemblies and water in the spent fuel pool or in the cask 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 $\leq$ 2430 pounds prior to moving it over fuel assemblies in the spent fuel pool or cask pit.

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.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool or cask 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 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.

ACTION:

a. With one emergency ventilation system servicing the storage pool area inoperable, fuel movement within the spent fuel pool or cask pit, or crane operation with loads over the spent fuel pool or cask 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 crane operation with loads over the spent fuel pool or cask 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 the criteria shown in Figure 3.9-1.

b. Fuel assemblies stored in the cask pit shall be placed in the spent fuel storage racks in accordance with the criteria shown in Figure 3.9-2.

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

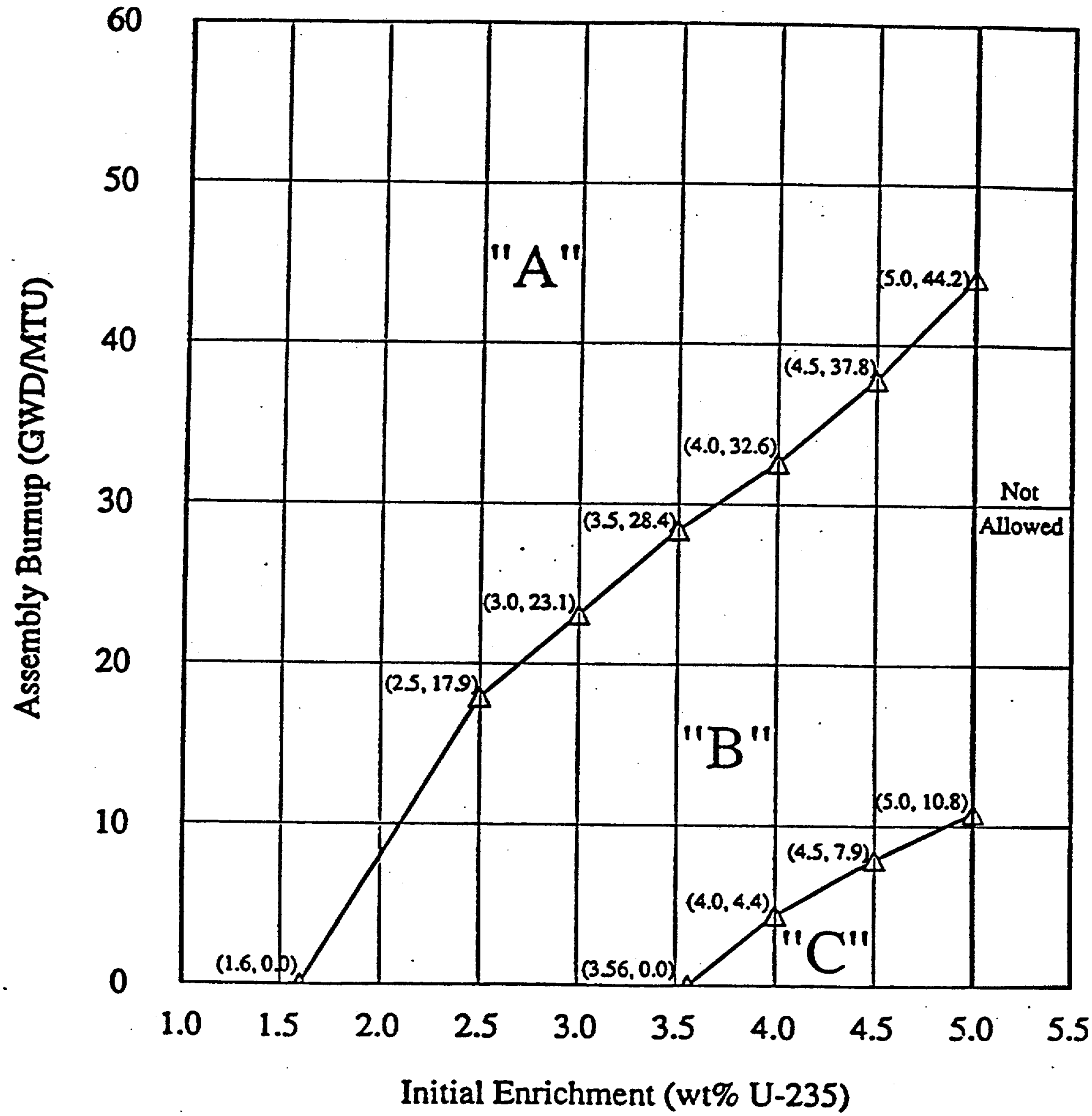
ACTION:

With the requirements of the above Specifications 3.9.13.a or 3.9.13.b not satisfied, suspend all other fuel movement within the spent fuel pool or cask 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 for the cask pit, as appropriate. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

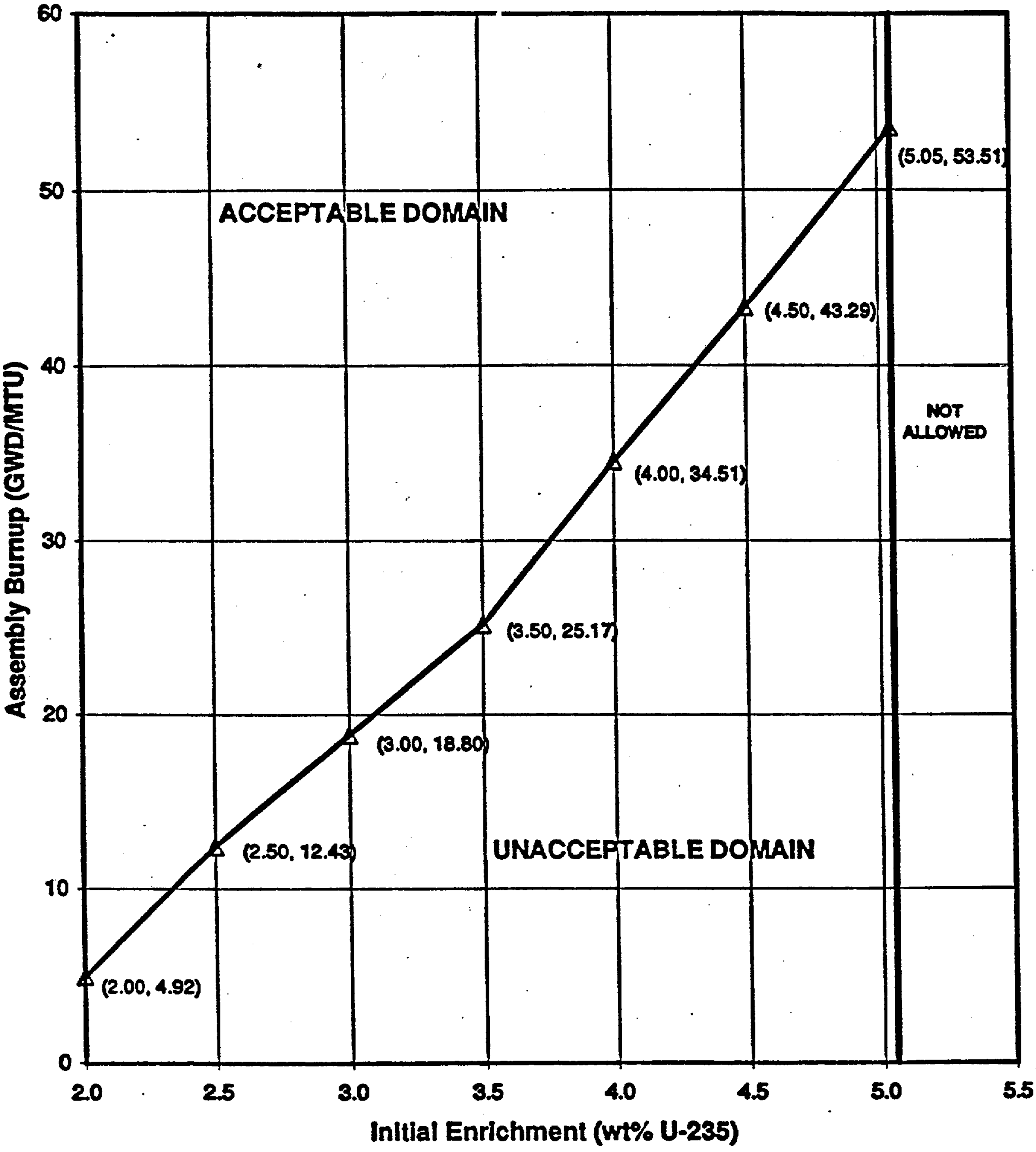
4.9.13.1 Prior to storing a fuel assembly in the spent fuel pool or cask pit, verify by administrative means that the initial enrichment and burnup of the fuel assembly are in accordance with Figure 3.9-1 for the spent fuel pool, or Figure 3.9-2 for the cask pit, as appropriate.

Figure 3.9-1
Burnup vs. Enrichment Curves For
Davis-Besse 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
Davis-Besse 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.

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 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.

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, as dictated by Figure 3.9-1 and Figure 3.9-2, ensure that the k-effective of the spent fuel pool and cask pit will always remain less than 0.95 assuming the spent fuel pool and cask pit to be flooded with non-borated water. The restrictions delineated in Figure 3.9-1 and Figure 3.9-2, and the action statement, are consistent with the criticality safety analyses performed for the spent fuel pool and cask pit.

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 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 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)

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 cask pit 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 for manufacturing tolerances and calculation uncertainty.

b. A rectangular array of stainless steel cells 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 cask pit in accordance with Technical Specification 3.9.13.

DESIGN FEATURES

5.6 Fuel Storage (continued)

5.6.2 Drainage

The spent fuel storage pool and cask 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 fuel assemblies.

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

5.7 (Deleted)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated May 21, 1999, and as supplemented by submittals dated December 1, 1999, and January 28, 2000, the licensee (FirstEnergy Nuclear Operating Company) of the Davis-Besse Nuclear Power Station Unit 1 (DBNPS) requested an amendment to Facility Operating License NPF-3. The amendment proposes changes to the DBNPS Technical Specifications (TSs) to allow 289 spent fuel assemblies (SFAs) to be temporarily stored in the cask pit (adjacent to the spent fuel pool).

The spent fuel pool (SFP) at the DBNPS has a storage capacity of 735 SFAs. As the result of the present unavailability of an off-site spent fuel storage facility and the spent fuel storage capacity restrictions at the site, the DBNPS is currently operating in Cycle 12 with insufficient[1] spent fuel storage capacity in the SFP to offload the entire core (177 fuel assemblies). A full core offload into the SFP is necessary for the performance of the ten-year in-service inspection required during the 12th refuel outage (RFO) in April 2000. In order to recover a full core offload capability, the licensee recently installed two rack modules[2] in the cask pit with a storage capacity of 153 SFAs. The addition of these two rack modules will allow the entire core to be offloaded for the ten-year in-service inspection during the 12th RFO and will also provide a full core offload capability during plant Cycle 13 operation.

The licensee has future plans to re-rack the SFP with maximum density racks. Two more rack modules with an additional storage capacity of 136 SFAs will be installed in the cask pit after the 12th RFO to provide temporary storage for shuffling of fuel during the future re-racking[3] of the SFP. All four of the cask pit storage racks will be relocated into the SFP as part of the final completion of the re-racking project.

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[1] Currently, only 114 empty spent fuel storage locations remain available in the SFP.

[2] These storage racks will remain unused until this license amendment request is approved by the staff.

[3] This license amendment request is only for the use of four storage racks in the DBNPS cask pit. Approval to re-rack the SFP with high density storage racks will be requested in a future license amendment submittal.

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original application.

2.0 EVALUATION

The proposed changes would expand the present spent fuel storage capability to allow the use of spent fuel racks in the cask pit area adjacent to the SFP. The SFP and cask pit are located within the fuel handling and storage area of the auxiliary building. The cask pit is independent of, and separate from, the SFP. It is used to transfer spent fuel to shipping casks or dry storage canisters and is accessible from the SFP through a 36-inch-wide opening in the 3-foot thick concrete wall dividing the two areas. The expansion will include four rack modules in the cask pit, increasing the available spent fuel storage locations by 289 cells.

The following Technical Specification (TS) changes have been proposed:

(1) TS 3/4.9.7, "Refueling Operations - Crane Travel - Fuel Handling Building," and related Bases would be revised to provide for storage of fuel assemblies in the cask pit.

(2) TS 3/4.9.11, "Refueling Operations - Storage Pool Water Level," would be revised to provide for storage of fuel assemblies in the cask pit.

(3) TS 3/4.9.12 , "Refueling Operations - Storage Pool Ventilation," would be revised to provide for storage of fuel assemblies in the cask pit.

(4) TS 3/4.9.13, "Refueling Operations - Spent Fuel Pool Fuel Assembly Storage," and related Bases would be revised to reflect the conditions required for fuel assembly storage in the cask pit area. The associated Figure 3.9-2, which shows the minimum required fuel assembly burnup as a function of initial enrichment for allowed storage in the cask pit racks, would be added.

(5) TS 5.6.1.1, "Design Features - Fuel Storage," would be revised to clarify that it applies to the spent fuel racks located in the spent fuel pool. TS 5.6.1.3 has been added to allow spent fuel storage of assemblies in the cask pit racks based on acceptable combinations of initial enrichment and discharge exposure as shown in TS Figure 3.9-2. TS 5.6.2 would be revised to reflect the storage of spent fuel assemblies in the cask pit. TS 5.6.3.b has been added to reflect the cask pit storage capacity limit of 289 fuel assemblies.

The staff's evaluation of the licensee's submittal focused on the following areas:

1 - Thermal Hydraulic Analyses
2 - Criticality Aspects of the Proposed Expansion of the Spent Fuel Pool
3 - Compatibility of Structural Materials and Boral with Spent Fuel Pool Environment
4 - Structural Integrity and Functionality of the Racks
5 - Control of Heavy Loads
6 - Dose Consequences of the Fuel Handling Accident
7 - Occupational Radiation Exposure
8 - Solid Radioactive Waste

2.1 THERMAL HYDRAULIC ANALYSES

In addition to the thermal-hydraulic analysis performed for the cask pit, the licensee also performed and presented in this submittal the thermal-hydraulic analysis for the SFP with the anticipation of the future SFP re-rack storage capacity.

2.1.1 Spent Fuel Pool Cooling

The spent fuel pool cooling and cleanup (SFPCC) system is designed to remove the decay heat from SFAs stored in the SFP, and to clarify and purify the water in the SFP. The SFP cooling portion of the SFPCC system consists of two half-capacity cooling trains each primarily equipped with one pump, one heat exchanger, and its associated valves, piping, instrumentation and controls. Heat is removed from the SFP heat exchanger by the component cooling water system. In addition, the decay heat removal (DHR) system which has a higher heat removal capacity, is used as a back-up system to cool the SFP when the reactor vessel is defueled.

The SFPCC system is designed to maintain the SFP water at or below 125^{0}F with both cooling trains operating to remove a partial core discharge (approximately 1/3 of a core) heat load of 12.4 x 10^6 Btu/hr. One DHR train alone is capable of maintaining the SFP water temperature at or below 147^{0}F with a decay heat load of 30.0 x 10^6 Btu/hr from an unplanned full core offload.

As indicated above, the licensee has long-term plans to re-rack the SFP with maximum density racks. The re-rack will increase the SFP storage capacity from 735 SFAs to approximately 1650 SFAs, which will result in an increase of SFP decay heat load for any specific core offload scenario. To account for the future re-racking of the SFP, the licensee reevaluated the thermal-hydraulic analyses conservatively based on the higher number of SFAs to evaluate the effects of the increase of SFA storage capacity on the SFPCC system and SFP water temperatures. The following summarizes the SFP peak temperatures and their corresponding heat loads in the SFP for the scenarios of a planned partial core offload and an unplanned full core offload:

Case	In-core Hold Time (hrs)	Max. SFP Temp. (°F)	Coincident[4] Time (Hrs after shutdown)	Coincident Hx Loads (MBtu/Hr)	SFP Boiling[5]	
					Time-to-Boil (hrs)	Boil-off rate (gpm)
1	150	132.98	183	15.98	10.42	34.45
2	150	151.42	203	29.75	3.78	69.57

Case 1[6]: Planned Partial Core Offload Refueling - Based on two years of full power operation, 1609 SFAs stored in the SFP prior to a final (last cycle) offload of 72 SFAs being discharged at a rate of four SFAs per hour into the SFP 150 hours after reactor shutdown, and cooling the SFP using two SFPCC pumps and two heat exchangers.

Case 2: Unplanned Full Core Offload - Based on 65 days of full power operation since the last planned partial core offload, 1537 SFAs stored in the SFP prior to a full core of 177 SFAs being discharged at a rate of four SFAs per hour into the SFP 150 hours after reactor shutdown, and cooling the SFP using one RHR system train.

As indicated in the above table, for Case 1, planned partial core offload scenario with the SFPCC system (both cooling trains are operating with one DHR system train as back-up) operating, the calculated peak SFP temperature resulting from the future re-racking SFP storage capacity is 132.98°F, which is still below the Standard Review Plan (SRP) Section 9.1.3, "Spent Fuel Pool Cooling And Cleanup System," temperature limit of 140°F for the SFP. For Case 2, unplanned full core offload scenario with one train of DHR system operating, the calculated peak SFP temperature is 151.42°F which is below the SRP guidance for the SFP water temperature limit (pool boiling) during unplanned full core offload outages.

Based on our review of the licensee's thermal-hydraulic analysis including the assumptions used to calculate decay heat loads in the SFP, we find the calculated peak water temperatures in the SFP during a planned partial core offload refueling or unplanned full core offload outages resulting from the future SFA storage capacity of not more than 1650 SFAs at the DBNPS acceptable.

Although the licensee had performed the above thermal-hydraulic analyses for the two discharge scenarios (planned partial core offload and unplanned full core offload) in accordance with the guidance as described in the SRP Section 9.1.3, the license also performed additional thermal-hydraulic analyses for various core discharge scenarios to demonstrate that the SFP

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[4] The time after reactor shutdown at which the SFP water reaches its calculated peak temperature.

[5] Resulting from an unlikely event of a complete loss of SFP cooling.

[6] DBNPS does not routinely conduct a full core offload during planned refueling outages.

water temperature will remain below boiling even under extreme circumstances. The following summarizes the heat loads and their corresponding peak temperatures in the SFP for various core discharge scenarios:

Case	In-core Hold Time (hrs)	Max. SFP Temp. (^{0}F)	Coincident Time (Hrs after shutdown)	Coincident Hx Loads (MBtu/Hr)
3	150	169.32	197	15.55
4	150	165.87	205	29.66
5	150	164.90	205	29.28
6	150	150.67	203	29.38

Case 3: Planned Partial Core Offload - Based on two years of full power operation, 1609 SFAs stored in the SFP prior to a final (last cycle) offload of 72 SFAs being discharged at a rate of four SFAs per hour into the SFP 150 hours after reactor shutdown, and cooling the SFP using one SFPCC pump and one heat exchanger.

Case 4: Unplanned Full Core Offload - Based on 65 days of full power operation since the last planned partial core offload, 1537 SFAs stored in the SFP prior to a full core of 177 SFAs being discharged at a rate of four SFAs per hour into the SFP 150 hours after reactor shutdown, and cooling the SFP using the SFPCC system (two pumps and two heat exchangers).

Case 5: Planned Full Core Offload - Based on two years of full power operation since the last planned partial core offload, 1537 SFAs stored in the SFP prior to a full core of 177 SFAs being discharged at a rate of four SFAs per hour into the SFP 150 hours after reactor shutdown, and cooling the SFP using the SFPCC system (two pumps and two heat exchangers).

Case 6: Planned Full Core Offload - Based on two years of full power operation since the last planned partial core offload, 1537 SFAs stored in the SFP prior to a full core of 177 SFAs being discharged at a rate of four SFAs per hour into the SFP 150 hours after reactor shutdown, and cooling the SFP using one DHR system train.

The SFP has a water temperature monitor system which alarms in the control room when the SFP water temperature reaches 125^{0}F. In the event that the alarm goes off due to high SFP temperature, plant procedures provide operator guidance to take corrective actions (i.e., to start the second SFPCC pump if only one pump is running, to utilize the DHR system, stop fuel movement, etc.). This will provide additional measures to prevent the SFP water temperature limits from being exceeded.

2.1.2 Cask Pit Cooling

With SFAs stored in the cask pit, natural circulation of SFP water will be relied upon to maintain water temperature limits within the cask pit. The cask pit is connected to the SFP by a gate that is isolated except during partial core or full core offload operations or with SFAs stored in the cask pit. The licensee stated that the temperatures between the cask pit and the SFP will equalize within a few degrees when the gate is removed and natural circulation between the regions occurs[7].

The license performed an analysis to evaluate the impact of the SFAs stored in the cask pit on the cask pit water temperatures without forced circulation cooling. The evaluation of buoyancy driven natural convection water exchange between the cask pit and the SFP yielded a maximum temperature difference of 4^{0}F. Therefore, the maximum temperatures in the cask pit would be approximately 137^{0}F for the scenario of planned partial core offload refueling outages and 155.5^{0}F for the scenario of unplanned full core offload outages. These temperatures are well below the guidance of SRP Section 9.1.3 for SFP water temperature limits.

In the analysis to evaluate the water temperature in the cask pit, relatively old SFAs were assumed to be stored in the cask pit. In the response to the staff's Request For Additional Information (RAI), the license stated that there are administrative controls established to ensure that each SFA must meet the following three restrictions in order to be stored in the cask pit:

(1) For criticality consideration, the SFA must meet a minimum burnup restriction which is a function of initial enrichment as specified in the proposed TS Figure 3.9-2, "Burnup vs. Enrichment Curve for Davis-Besse Cask Pit Storage Racks," before the SFA will be allowed to be stored in the cask pit storage racks.

(2) For thermal-hydraulic consideration, the SFA must not exceed a maximum heat generation rate restriction of 873 watts. A SFA will not be allowed to be stored in the cask pit storage racks if this maximum heat generation rate is exceeded.

(3) For radiological consideration, the SFA must meet a restriction of 3 years minimum decay time since last irradiation. A SFA will not be allowed to be stored in the cask pit storage racks if this irradiation restriction is not met.

Together, the above three restrictions will prevent SFAs from a recently offloaded core from being placed in the cask pit fuel racks. In addition, there are procedures established to ensure that the gate between the cask pit and the SFP will be removed prior to storage of SFAs in the cask pit.

Based on our review of licensee's assumptions used to calculate the decay heat loads in the cask pit and the license's procedures described above, we conclude that the peak calculated water temperatures in the cask pit during planned partial core offload refueling or unplanned full

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[7] Decay heat from the SFAs placed in the cask pit will heat up the water causing it to flow to cooler regions (the SFP) where it will be removed by the SFPCC system.

core offload outages resulting from placing SFAs in the cask pit and the increase of SFA storage capacity at the DBNPS are acceptable.

2.1.3 Effects of SFP Boiling

In the unlikely event that there is a complete loss of cooling, the SFP water temperature will begin to rise and will eventually reach pool boiling. Taking the future re-racking into account and based on the most severe scenario (unplanned full core offload), the minimum time from the loss-of-pool-cooling at peak pool water temperature until the pool boils is 3.78 hours with a maximum boil-off rate of 69.57 gpm. The licensee stated that there is sufficient time to provide make-up via a variety of proceduralized valve line-ups including gravity fill methods using the borated water storage tank, the demineralized water storage tank, or the clean waste receiver tanks as water sources. In the unlikely event that the establishment of make-up to the SFP was delayed following 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 SFAs to the level corresponding to 9½[8] feet above the top of SFAs stored in the racks.

Based on our review of the licensee's evaluation of the impact of a complete loss of SFP cooling, we find that the licensee has sufficient time and is capable of aligning make-up water to the pool before boiling begins and that make-up water will be supplied at a rate which exceeds the boil off rate. We conclude that cooling of the SFP at the DBNPS conforms with the guidance as described in the SRP and is, therefore, acceptable.

2.1.4 Fuel Handling Area Ventilation

The fuel handling area ventilation system is designed to maintain the fuel handling area between 60°F and 110°F. The licensee performed an analysis to evaluate the impact of the SFAs to be stored in the cask pit and the increase of SFA storage capacity in the SFP on the fuel handling area ventilation system. The licensee stated that, based on the most limiting full core offload scenario, the maximum calculated building temperature is 103°F, which is within the design capability of the fuel handling area ventilation system.

Based on our review of the licensee's analysis and the experience gained from our review of SFP re-rack applications for other facilities, we conclude that the additional storage of SFAs in the cask pit and SFP will have an insignificant impact on the fuel handling area ventilation system.

2.2 CRITICALITY ASPECTS OF THE PROPOSED EXPANSION OF THE SPENT FUEL POOL

The analysis of the reactivity effects of fuel storage in the spent fuel racks was performed with the continuous energy three-dimensional Monte Carlo code, MCNP4a. Independent verification calculations were performed with KENO5a, a three-dimensional multigroup Monte Carlo code,

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8 A minimum of 9½ feet of borated water above the top of the active fuel stored in the racks is required to ensure adequate biological shielding.

using the 238-group SCALE-4.3 cross-section library. Fuel depletion analyses were made with CASMO-4, a two-dimensional multigroup transport theory code. The determination of small reactivity increments due to manufacturing tolerances were also made with CASMO-4. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous criticality experiments. These experiments simulate the DBNPS spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and B-10 loading in the absorber. The two independent methods of analysis (KENO5a and MCNP4a) showed good agreement both with experiment and with each other. The comparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO5a calculations, a minimum of 1,000,000 neutron histories were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of MCNP4a and KENO5a reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the DBNPS storage racks with a high degree of confidence.

The cask pit is normally flooded with water borated to at least 1800 parts per million (ppm) of boron, which results in a large subcriticality margin under actual operating conditions. However, to assure the criticality safety under normal and accident conditions and to conform to the requirements of General Design Criterion (GDC) 62 for the prevention of criticality in fuel storage and handling, the criterion stated in SRP Section 9.1.2, "Spent Fuel Storage," must be satisfied. This criterion states that the maximum reactivity (k_{eff})of the racks containing fuel of the highest anticipated reactivity and flooded with unborated water must not exceed 0.95. The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed 0.95 at a 95 percent probability, 95 percent confidence (95/95) level.

The criticality analyses were performed with several conservative assumptions which tend to maximize the rack reactivity in the cask pit. These include:

(1) unborated water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures,

(2) an infinite array of storage cells in the lateral direction (except for the assessment of peripheral effects and certain abnormal conditions where neutron leakage is inherent),

(3) neutron absorption effect of minor structural material is neglected,

(4) racks are fully loaded with the most reactive fuel authorized to be stored at the DBNPS without any control rods or burnable poison,

(5) no credit for the water gap between the racks or the additional Boral panel between adjacent racks.

The design basis fuel assembly is the most reactive (minimum cladding thickness) Babcock & Wilcox Mark B assembly with a 15x15 array of fuel rods containing UO_2 at a maximum initial enrichment of 5.0 + 0.05 w/o U-235 (manufacturing tolerance) with 17 fuel rods replaced by 16 control rod guide tubes and one instrument tube.

The staff concludes that appropriately conservative assumptions were made.

For the analysis of the cask pit storage configuration, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, fuel enrichment and density tolerances, water-gap spacing between modules, and eccentric fuel positioning were accounted for. These uncertainties were appropriately determined at the 95/95 probability/confidence level. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations and the effect of the axial distribution in burnup.

CASMO-4 was used for in-core depletion calculations in the hot operating condition. The highest fuel and moderator temperature (1300 °F and 610 °F, respectively), a soluble boron concentration of 1000 ppm, and burnable poison (4.0 w/o B_4C) rods present in each guide tube (removed at 35,000 MWD/MTU) were assumed. These assumptions assure the highest plutonium production and, hence, conservatively high values of reactivity during burnup. Reactivity equivalencing calculations were made for fuel of several different initial enrichments and interpolated to define the burnup-dependent equivalent enrichments that yield the same reactivity ($k_{eff} \leq 0.95$) in the storage rack configuration, as shown in TS Figure 3.9-2. This figure shows that the cask pit storage racks can safely accommodate fuel of various initial enrichments up to 5.05 w/o and discharge burnups provided the combination falls within the acceptable domain illustrated by the upper line. This reactivity equivalencing method is the standard one used for storage rack reactivity evaluations and is acceptable.

The final maximum calculated reactivity resulted in a k_{eff} of 0.9452 for 5.05 w/o U-235 fuel at a burnup of 53,510 megawatt days per metric ton uranium (MWD/MTU), when combined with all known uncertainties. This value meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level. Therefore, the proposed storage of fuel enriched to 5.05 w/o U-235 in the racks installed in the cask pit is acceptable if it meets the burnup versus initial enrichment curve given in TS Figure 3.9-2.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable area in TS Figure 3.9-2, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of at least 1800 ppm of boron in the pool water, which is administratively controlled during and following fuel movement, until completion of verification that no misloading has occurred. This is allowed because the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident. The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. In fact, the licensee has confirmed that a minimum boron concentration of only 650 ppm boron would be adequate to assure that the limiting k_{eff} of 0.95 is not exceeded for the worst accident.

Based on the review described above, the staff finds the criticality aspects of the installation of new spent fuel storage racks in the cask pit area adjacent to the DBNPS spent fuel pool are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling.

2.3 COMPATIBILITY OF STRUCTURAL MATERIALS AND BORAL WITH SPENT FUEL POOL ENVIRONMENT

The new storage rack arrays proposed for use in the spent fuel pool and cask pit area are manufactured by Holtec International. These free-standing, self-supporting racks are designed to stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel Code. They are made primarily from ASME Type 304 austenitic stainless steel containing honeycomb storage cells interconnected through longitudinal welds.

2.3.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-304 for all sheet metal, basemetal, and cell connecting bar stock and internally threaded support legs, ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100 °F) for externally threaded support spindle, and ASME Type 308 and ASME Type 308L for weld material.

These materials used in the Holtec racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the spent fuel pool environment. Therefore, they are acceptable for use in this application.

2.3.2 Poison Material

The Holtec racks employ Boral™ as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the spent fuel pool environments where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation, and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed with spot welding to vent the corrosion gases. The neutron absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

Based on its evaluation, the staff finds the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec International are compatible with the spent fuel pool environment at the DBNPS. The type of degradation exhibited by the racks does not affect their neutron absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

2.4 STRUCTURAL INTEGRITY AND FUNCTIONALITY OF THE RACKS

The primary purpose of this review is to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the cask pit structure subject to the effects of the postulated loads (Appendix D of SRP Section 3.8.4, "Other Seismic Category I Structures") and fuel handling accidents.

2.4.1 Storage Racks

The licensee has proposed to install four racks in the cask pit. All four storage racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee, along with its contractor, Holtec, performed structural analyses of the racks for the requested license amendment.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the DBNPS spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and are not attached to the floor or walls of the cask pit. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the program, were used to simulate the three dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Analyses of two models were performed: a 3-D single rack (SR) model and a 3-D multi-rack (MR) model. For the 3-D SR analyses, the rack was considered to be fully loaded and half loaded with three different coefficients of friction (μ=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the cask pit floor were used to investigate the stability of the rack with respect to overturning. For the 3-D MR analyses, all racks were considered to be fully loaded, partially loaded and almost empty with three different coefficients of friction (μ=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the cask pit floor. The analyses were performed to investigate the fluid-structure interaction effects between the racks and the cask pit walls as well as those among the racks and to identify the worst case response for rack movement and for rack member stresses.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) were generated from the design response spectra defined in the DBNPS Updated Safety Analysis Report (USAR). The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density function compatible with the design response spectra as discussed in SRP Section 3.7.1, "Seismic Design Parameters."

A total of thirty (30) 3-D SR and MR analyses were performed. The racks were subjected to the service, upset and faulted loading conditions (Level A, B and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 0.445 inch indicating that there is adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the rack and the cask pit wall. However, the results show that there is impact potential between the racks. The staff compared the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The stress results show that the induced impact forces under the SSE loading condition are small and all induced stresses in the racks are smaller than the corresponding allowable stresses specified in the ASME Boiler and Pressure Vessel Code indicating that the rack design is adequate.

The licensee also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code indicating that the weld connection design of the rack is adequate.

Based on: (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, and (3) the licensee's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated design loading conditions.

2.4.2 Cask Pit Structure

The licensee analyzed the cask pit structure to demonstrate the adequacy of the structure under fully-loaded fuel racks with all storage locations occupied by fuel assemblies. The fully-loaded structures were subjected to the load combinations specified in the DBNPS USAR.

The licensee's analysis shows the predicted minimum factors of safety varying from 1.41 to 43.81 for shear force and bending moment of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading and SSE loading conditions. Thus, the cask pit structural design is acceptable.

2.4.3 Fuel Handling Accident

The following two refueling accident cases were evaluated by the licensee: (1) drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario) and (2) drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of accident case (1) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area. Therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of accident drop case (2) show that damage will be restricted to a depth of 4.75 inches below the top of the rack, which is above the active fuel region. The staff reviewed the licensee's analysis results and concurs with its findings. This is acceptable based on the licensee's structural integrity conclusions supported by the parametric studies.

Based on the review and evaluation of the licensee's submittal, the staff concludes that the structural analysis and design of the spent fuel rack modules and the cask pit structure are adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with the current licensing basis set forth in the DBNPS USAR and applicable provisions of the SRP and are, therefore, acceptable.

2.5 CONTROL OF HEAVY LOADS

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines for licensees to assure safe handling of heavy loads by prohibiting load travel, to the extent practicable, over spent fuel assemblies, over the core, and over safety-related equipment. It also defines a heavy load as the combined weight of a single fuel assembly and its handling tool. Furthermore, it recommends in Section 5.1.1 that procedures be developed to cover load handling operations for heavy loads that could be handled over or in proximity to irradiated fuel.

The current DBNPS TS restricts loads that exceed 2430 pounds (heavy loads) from travel over fuel assemblies stored in the SFP. The licensee is proposing to apply the current TS restrictions to spent fuel stored in the cask pit. Additionally, defense-in-depth guidelines in accordance with NUREG-0612 will be used to assure that heavy loads are not moved over fuel in the SFP or the cask pit.

2.5.1 TS 3/4.9.7, "Refueling Operations - Crane Travel - Fuel Handling Building" and associated Bases

Existing TS Limiting Condition for Operation (LCO) 3.9.7 and associated Bases prohibits movement of loads greater than 2430 pounds (heavy loads) over fuel assemblies stored in the SFP. The proposed TS LCO will prohibit the movement of heavy loads over fuel assemblies stored in either the SFP or the cask pit. LCO 3.9.7 Applicability Statement, Surveillance Requirement 4.9.7, and Bases 3/4.9.7 will also be revised to allow storage of spent fuel assemblies in the cask pit.

This TS limit is consistent with the design basis fuel handling accident and was established to (1) limit any release outside containment due to a dropped load damaging all the fuel rods in a single fuel assembly and (2) avoid any possible distortion of fuel in the storage racks that may result in a critical array. The weight limit on loads that could be moved over fuel stored in the cask pit and the SFP enables the licensee to maintain safety consistent with the guidelines in NUREG-0612, and, therefore, is acceptable.

2.5.2 Surveillance Requirement (SR) 4.9.7

SR 4.9.7 will be changed to require that, prior to moving any loads other than a fuel assembly over irradiated fuel in the SFP or cask pit, the licensee verifies that the loads are $\leq$ 2430 pounds. This proposed change will ensure that loads moved over fuel in the cask pit are within the safety limits established in the fuel handling accident analysis under USAR Section 15.4.7 "Fuel Handling Accident."

2.5.3 Hoisting System

The Spent Fuel Cask Crane (SFCC) will be used in conjunction with a temporary hoist for installation of the new storage racks in the cask pit. The crane has a capacity of 140 tons in the main hoist and 20 tons in the auxiliary hoist. As stated by the licensee, they will ensure that the crane is designed in accordance with the requirements of "Specification No. 70 for Electric Overhead Traveling Cranes," of the Crane Manufacturers Association of America, and ANSI B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)." The maximum lift weight during the rack installation is 15,650 lbs., which includes the rack, lift rig, rigging and the temporary hoist. Therefore, the crane capacity for the lift provides a large factor of safety.

The licensee states that a remotely engaging lifting rig will be interposed between the crane hook and the rack. The lifting rig is specifically designed to lift the new spent fuel rack modules. It is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." It consists of four independently loaded lift rods and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. As stated by the licensee, both the stress design and the load testing of the lifting rig satisfies guidelines in Section 5.1.6(1) of NUREG-0612 and ANSI N14.6 (1978), respectively. Accordingly, the lift rods are designed as follows: (1) with the appropriate stress design factor as specified in ANSI N14.6 (safety factor of 5 to 1); (2) load tested to 300% of the maximum weight to be lifted; and (3) after load testing, the integrity of the critical weld joints is examined using a liquid penetrant. Non-customized lifting devices (i.e., slings) will be used in accordance with NUREG-0612 and ANSI B30.9-1971, "Slings." Therefore, the slings must be proof tested at a minimum of 1.5 times their rated capacity in accordance with Section 9.3.3 in ANSI B30.9.

The staff believes that the crane coupled with the design and testing of the lifting rig and other lifting devices will enable the licensee to handle heavy loads with little to no risks to the safety of the rerack operation.

2.5.4 Load Handling Accident Analysis

The licensee considered a load handling accident involving a dropped spent fuel rack into the cask pit and found that a dropped rack would not lead to catastrophic leakage of the cask pit. Although a dropped rack may cause cracks to develop throughout the cask pit floor, guidelines in NUREG-0612 will be used to preclude a rack drop. A safe load path will be established to avoid rack lifts over the SFP or any safety-related equipment. The SFCC is interlocked to avoid travel over the SFP; therefore, the new racks will not be lifted over any portion of the SFP.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee plans to implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. They will provide: (1) comprehensive training to the rerack installation crew, (2) use redundantly designed lifting rigs, (3) perform inspection and maintenance checks on the cranes and lifting devices prior to the rerack operation, and (4) use specific procedures that cover the entire rerack effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement.

The staff agrees with the licensee that the use of the crane in conjunction with administrative procedures and controls focused on, but not limited to, the areas noted above will enable the licensee to maintain safety during the rerack operation.

Based on the preceding discussions, the staff finds that the proposed changes to the TS to support storage of spent fuel in the cask pit and the use of administrative controls to improve the handling of the racks in the cask pit are in accordance with NUREG-0612. These changes will enable the licensee to move the racks while preventing any damage to spent fuel, the SFP, and the cask pit if a failure were to occur.

2.6 DOSE CONSEQUENCES OF THE FUEL HANDLING ACCIDENT

The staff reviewed the proposal with regard to the dose consequences of the design basis fuel handling accident (FHA). The licensee proposes to restrict fuel to be placed in the cask pit racks to fuel that has been removed from the reactor for at least three years. The length of the decay period was determined by the licensee to address on-site ALARA (as low as reasonably achievable) and thermal-hydraulic considerations. The licensee will establish administrative controls to ensure the three-year age limitation will not be violated.

The licensee's current evaluation of the consequences of the FHA outside containment is provided in Section 15.4.7.2 of the DBNPS USAR. This analysis assumes the entire outside row of fuel rods (56 of 208) fail in a fuel assembly that has undergone 72 hours of decay time. The dose consequences are well within limits established in 10 CFR Part 100; therefore, meeting the acceptance criteria of SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," (i.e., 75 rem thyroid and 6 rem whole body). The licensee performed a calculation assuming all rods fail in a fuel assembly that has undergone three years of decay to show that the dose consequences from an FHA in the cask pit are bounded by the current USAR FHA outside containment dose analysis results. The licensee concluded that the radiological dose from an FHA in the cask pit would not be increased from that previously considered.

The storage of spent fuel in racks in the cask pit will not cause changes to any of the FHA dose analysis inputs or assumptions other than the radiological decay period. The staff performed a confirmatory calculation with the longer decay period and agrees with the licensee's conclusion that the current analysis is bounding. Since the dose consequences of the current FHA outside containment dose analysis meet the SRP acceptance criteria and also bound the dose from an FHA in the cask pit, the staff finds the radiological consequences of storing spent fuel assemblies in the cask pit acceptable.

2.7 OCCUPATIONAL RADIATION EXPOSURE

The staff has reviewed the licensee's plan for the installation of additional spent fuel rack modules at the DBNPS with respect to occupational radiation exposure. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed for between 1.85 and 4 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA principles. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained ALARA. Personnel will wear protective clothing and respiratory protective equipment, if necessary. The licensee will issue personnel monitoring equipment to each individual.

The licensee will use divers for the installation of the four fuel rack modules in the cask pit. Each diver will be equipped with whole body and extremity dosimetry with remote, above surface readouts which will be continuously monitored by Radiation Protection personnel. The divers will also be equipped with underwater survey instrumentation with remote readout capabilities. In addition, the divers will be in continuous communication with Radiation Protection personnel. The licensee will conduct radiation surveys of the diving area prior to each diving operation and following the movement on any irradiated hardware in the cask pit. The licensee 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 licensee will also use a safety line attached to the diver and manned by a dive tender at all times to maintain positive diver control.

The licensee will install the fuel rack modules in the cask pit in two separate phases. The first two rack modules have previously been installed in the cask pit and are intended for use during the spring refueling outage. The second two fuel rack modules will be installed following the outage. At this time, any fuel assemblies that had been stored in the cask pit will be transferred to the SFP. In addition, the licensee will rearrange spent fuel assemblies stored in the adjacent SFP to minimize their dose contribution to divers working in the cask pit. The licensee will use an underwater vacuum cleaner system to remove any crud or debris from the bottom of the cask pit prior to installation of the remaining two rack modules.

After each installation of the fuel rack modules in the cask pit, the licensee will wash the lifting device with demineralized water and wrap it to control the spread of contamination. The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase as a result of the expanded spent fuel storage capacity. However, the licensee will operate continuous air monitors in areas where there is a potential for significant airborne activity during installation of the fuel rack modules in the cask pit.

Elevation 603' (pool surface level) of the Auxiliary Building in the vicinity of the SFP and cask pit has a radiation level Zone C (<15 mrem/hr) designation. The licensee estimates that storage of spent fuel in the cask pit will not result in dose rates above the cask pit that will exceed the current radiation zone designation for this area (the licensee will administratively limit fuel that can be stored in the cask pit to spent fuel that has been out of the core for at least three years). The two rooms (Rooms 106 and 109) which are located adjacent to the cask pit on a lower elevation have maximum dose rate designations of 100 mrem/hr and 1000 mrem/hr, respectively. The licensee has calculated that the storage of three-year decayed fuel in the cask pit will result in a dose rate of approximately 70 mrem/hr near the ceilings of these rooms. Therefore, the storage of spent fuel in the cask pit will not result in any change in the dose rate designations for these two adjacent rooms.

On the basis of our review of the proposed license amendment, the staff concludes that the proposed increase in spent fuel storage capacity at the DBNPS can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds that the projected dose for the project of 1.85 to 4 person-rem is in the range of doses for similar modifications at other plants and is, therefore, acceptable.

2.8 SOLID RADIOACTIVE WASTE

Spent resins are generated by the processing of SFP water through the SFP Purification System. These spent resins are changed out about once every 18 months and the licensee does not expect this change-out frequency to be significantly affected by the storage of spent fuel in the cask pit. In order to maintain the cask pit water as clean as possible, and thereby minimizing the generation of spent resins, the licensee will vacuum the floor of the cask pit to remove any radioactive crud, sediment, and other debris before the new fuel rack modules are installed. Filters from use of this underwater vacuum system will be a source of solid radwaste. Overall, however, the licensee does not expect that the storage of spent fuel in the cask pit will result in a significant change in the generation of solid radwaste at the DBNPS.

On the basis of our review of the proposed license amendment, the staff concludes that the proposed increase in spent fuel storage capacity at the DBNPS can be performed in a manner that will ensure that doses to the workers will be maintained ALARA and the generation of additional solid radioactive wastes minimized.

2.9 SUMMARY

The staff has performed a detailed review of the licensee's proposal that would permit them to store spent fuel assemblies in the cask pit area adjacent to the SFP. The staff's review ensured that (1) the heat transfer capability of the SFP and cask pit area would be sufficient to handle the increased number of stored fuel assemblies; (2) there would not be an inadvertent criticality associated with the increased number of stored fuel assemblies; (3) the materials of the new racks are compatible with the spent fuel assemblies and the spent fuel pool environment; (4) the structural analysis and design of the spent fuel rack modules and the cask pit structure are adequate to withstand the effects of the applicable loads; (5) the licensee will adequately control the movement of heavy loads over the SFP and cask pit area; (6) the dose consequences of the fuel handling accident remain within the established licensing basis; (7) the occupational dose consequences remain with the established licensing basis; and (8) the generation of additional solid radioactive wastes will be minimized.

On the basis of the staff's review, the staff finds the proposed modifications acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. Comments provided by the State official, along with the staff's response, are provided below.

The cask pit is designed to support movement of spent fuel assemblies prior to their being placed in onsite dry fuel storage or shipped offsite. As described in DBNPS USAR Section 9.1.4.3, a spent fuel transfer cask is placed in the cask pit for transfer of spent fuel assemblies from the spent fuel pool. Certificate of Compliance No. 1004, which authorizes onsite dry fuel storage, was originally issued to VECTRA and is now possessed by Transnuclear West, Incorporated, vendor of the NUHOMS-24P horizontal storage module (HSM). The DBNPS currently has three HSMs. Each HSM contains 24 spent fuel assemblies for a total of 72 assemblies in dry storage.

During the licensing for onsite dry fuel storage, the licensee was required to develop procedures for unloading an HSM if conditions are warranted. The Certificate of Compliance requires daily surveillances to verify that the HSM air inlets and outlets are not blocked. A conservative analysis of complete blockage of all air inlets and outlets indicates that the concrete can reach the accident temperature limit of 350°F in approximately 40 hours. The Certificate of Compliance further states that if the concrete accident temperature criteria (350°F) has been exceeded for more than 24 hours, the HSM must be removed from service (i.e., the spent fuel assemblies must be removed). The licensee's procedures would require removing the dry shielded cannister (containing the spent fuel assemblies) from the concrete HSM, placing the dry shielded cannister into the transfer cask, and returning the transfer cask to the cask pit for removal of the spent fuel assemblies.

The State official questioned whether the cask pit would be available for unloading an HSM if the cask pit is being used to store spent fuel assemblies. The cask pit will store spent fuel assemblies during the April 2000 refueling outage. In addition, the cask pit will also be used for storing and shuffling spent fuel assemblies during the licensee's planned rerack of the spent fuel pool. Two additional fuel storage racks, which are scheduled to be fabricated in 2001, will be placed in the cask pit (for a total of four) to support the licensee's long term plan to rerack the spent fuel pool with high density fuel racks.

The staff considers the need to unload an HSM to be extremely small. The HSMs, which have been designed for a heat load capacity of 24 kW, have been loaded with fuel assemblies representing a total heat load of approximately 12 kW. Therefore, the HSMs are fully expected to handle the existing heat loads of the spent fuel assemblies. In the three years that the licensee has maintained dry fuel storage, the maximum concrete temperatures of the concrete modules has been approximately 120°F. The licensee is required to monitor HSM temperatures on a daily basis and would have sufficient time to monitor and take actions if temperatures began to trend upwards.

The Certificate of Compliance for the NUHOMS HSM does not include time constraints for removing an HSM from service. The absence of time constraints provides the licensee the opportunity to consider alternative actions as well as acknowledging that considerable time may be required to perform such actions. If conditions warrant removing spent fuel assemblies from an HSM, a number of activities would need to take place. Special equipment would need to be brought onsite as well as the appropriate contractors. The cask pit would need to be cleared of all spent fuel assemblies as well as the fuel storage racks. Clearing the cask pit may also require returning spent fuel assemblies and the reactor vessel internals back to the reactor vessel. Considering that the licensee has scheduled the entire core offload and ISI inspections for a nine day window, the staff believes that conditions would justify completion of the ISI inspections and refueling activities prior to clearing the cask pit and unloading an HSM.

The State official also expressed concern about the possibility of a transfer cask being inadvertently dropped on spent fuel assemblies in the cask pit. The staff pointed out that DBNPS Technical Specification 3.9.7, "Refueling Operations - Crane Travel - Fuel Handling Building," is being revised as part of this license amendment to prohibit loads in excess of 2430 pounds from traveling over fuel assemblies in either the spent fuel pool or the cask pit. A transfer cask containing a fully loaded dry shielded cannister is estimated to weigh approximately 100 tons. The staff's safety evaluation addresses the licensee's compliance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," to ensure safe handling of heavy loads.

In summary, the staff concludes that the licensee can fully support storage of spent fuel assemblies in the cask pit. The licensee has adequate procedures to unload an HSM if necessary and is fully expected to take appropriate actions.

The State official had no further comments

4.0 <u>ENVIRONMENTAL CONSIDERATION</u>

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on January 13, 2000 (65 FR 2201), in connection with the proposed technical specification changes. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 <u>CONCLUSION</u>

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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