October 19, 2001

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

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. MB0688)

Dear Mr. Campbell:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 247 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit 1. The amendment revises the Technical Specifications (TSs) in response to your application dated December 2, 2000, as supplemented by letters dated September 4 and September 28, 2001.

This amendment increases the 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. The amendment also approves additional temporary storage of up to 90 fuel assemblies in the fuel transfer pit to support a complete re-racking of the SFP. The increase in SFP storage capacity will provide a full core offload capability during the plant's Cycle 13 operation and enable the Davis-Besse facility to meet its storage needs through April 22, 2017, which is the expiration date for the current operating license.

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,

***/Douglas V. Pickett for/***

Stephen P. Sands, 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. 247 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

Mr. Guy G. Campbell, Vice President - Nuclear October 19, 2001
FirstEnergy Nuclear Operating Company
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. MB0688)

Dear Mr. Campbell:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No.247 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit 1. The amendment revises the Technical Specifications (TSs) in response to your application dated December 2, 2000, as supplemented by letters dated September 4 and September 28, 2001.

This amendment increases the 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. The amendment also approves additional temporary storage of up to 90 fuel assemblies in the fuel transfer pit to support a complete re-racking of the SFP. The increase in SFP storage capacity will provide a full core offload capability during the plant's Cycle 13 operation and enable the Davis-Besse facility to meet its storage needs through April 22, 2017, which is the expiration date for the current operating license.

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,
***/Douglas Pickett for/***
Stephen P. Sands, 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. 247 to License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

Distribution w/encls:

| | | |
|---|---|---|
| PUBLIC | WBECKNER | KMANOLY |
| PDIII/2 | GGRANT, RIII | |
| AMENDIOLA | SSANDS | |
| THARRIS | ESULLIVAN | |
| OGC | KGIBSON | |
| ACRS | AKSTULEWICZ | |
| GHILL (2) | JHANNON | |

ADAMS ACCESSION NUMBER: **ML012910442**

| OFFICE | PM:PD3-2 | LA:PD3-2 | SC:EMCB | SC:IOLB | SC:SRXB |
|---|---|---|---|---|---|
| NAME | SSands/***DVP for/*** | THarris | ESullivan* | KGibson* | FAkstulewicz** |
| DATE | 10/19/2001 | 10/19/2001 | 3/7/2001 | 9/17/01 | 9/20/01 |

| OFFICE | BC:SPLB | SC:EMEB | | OGC | SC:PD3-2 |
|---|---|---|---|---|---|
| NAME | JHannon* | KManoly* | | CMarco | AMendiola |
| DATE | 10 / 17 /2001 | 10/ 16 /2001 | | 10/19/2001 | 10/19/2001 |

OFFICIAL RECORD COPY

*See memorandums to AMendiola
**See memoranda to SSands

cc:

Mary E. O'Reilly
FirstEnergy
76 South Main Street
Akron, OH 44308

Manager - Regulatory Affairs
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

Jay E. Silberg, Esq.
Shaw, Pittman, Potts
and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Regional Administrator
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60523-4351

Michael A. Schoppman
Framatome ANP
1911 N. Ft. Myer Drive
Rosslyn, VA 22209

Resident Inspector
U.S. Nuclear Regulatory Commission
5503 North State Route 2
Oak Harbor, OH 43449-9760

Plant Manager
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

Harvey B. Brugger, Supervisor
Radiological Assistance Section
Bureau of Radiation Protection
Ohio Department of Health
P.O. Box 118
Columbus, OH 43266-0118

Carol O'Claire, Chief, Radiological Branch
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-2206

Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH 43068-9009

Ohio Environmental Protection Agency
DERR--Compliance Unit
ATTN: Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Attorney General
Department of Attorney
30 East Broad Street
Columbus, OH 43216

President, Board of County
Commissioners of Ottawa County
Port Clinton, OH 43252

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 247
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 December 2, 2000, as supplemented by letters dated September 4 and September 28, 2001, 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. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 247, 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 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

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: October 19, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 247

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 |
|---|---|
| 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 | 3/4 9-15 |
| ----- | 3/4 9-16 |
| 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 | 5-3 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 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 December 2, 2000, and as supplemented by letters dated September 4 and September 28, 2001, the FirstEnergy Nuclear Operating Company (FENOC, or the licensee), proposed revisions to the Technical Specifications (TS) to support an increase in the spent fuel storage capacity at the Davis-Besse Nuclear Power Station (DBNPS), Unit 1. The proposed amendment requests to increase the current spent fuel storage capacity from 13 racks with 735 spent fuel assemblies (SFAs) to 21 racks with 1624 SFAs. To support the complete re-racking of the spent fuel pool (SFP), the proposed amendment also requests to install a rack in the transfer pit for additional temporary storage of fuel assemblies, if needed. Approval for up to 90 transfer pit storage locations is requested, and the rack in the transfer pit will be relocated to the SFP to complete the rerack project. Finally, the proposed amendment requests to relocate four previously installed racks from the cask pit to the SFP as part of the complete re-rack project.

The proposed increase in SFP storage capacity will provide a full core offload capability during the plant's Cycle 13 operation and enable DBNPS to meet its storage needs through April 22, 2017, which is the expiration date for the current operating license.

Specifically, in addition to proposing an increase in the SFP storage capacity, the amendment proposes to revise 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," to support the increase in SFP storage capacity.

The following evaluation covers the applicable areas of the licensee's submittal: thermal-hydraulic analysis of the SFP, cask pit, transfer pit; fuel handling area ventilation; control and handling of heavy loads; criticality analysis of the high density racks; structural materials analysis; structural mechanics analysis; and occupational radiation protection and radioactive waste.

The supplemental letters of September 4 and September 28, 2001, contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

## 2.0 THERMAL-HYDRAULIC ANALYSIS OF SFP, CASK PIT, AND TRANSFER PIT, FUEL HANDLING AREA VENTILATION, AND CONTROL AND HANDLING OF HEAVY LOADS

### 2.1 Applicable Regulatory Guidance

NUREG-0800, Standard Review Plan (SRP), Section 9.1.2 provides criteria for staff review of the design and performance of the spent fuel storage facility, including the spent fuel storage racks, the SFP, the SFP liner plate, and associated storage pits to assure conformance with the requirements in General Design Criteria (GDC) 2, 4, 5, 61, 62, and 63. Accordingly, the staff reviews whether (1) the safety function of the spent fuel pool and storage racks is maintained; (2) the spent fuel assemblies are in a safe and sub-critical array during all credible storage conditions; and (3) a safe means of loading the assemblies into storage and shipping casks is provided. Section 9.1.3 in NUREG-0800 provides criteria for staff review of the design and performance of SFP cooling and cleanup (SFPCC), including the adequacy of the SFPCC system, and the fuel pool make-up water systems to assure conformance with the requirements in GDC 2, 4, 5, 44, 45, 61, 63, and 10 CFR Part 20.

Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" provides methods acceptable for the licensee to implement GDC 61 of Appendix A to 10 CFR Part 50. GDC 61 requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions.

Nuclear Regulatory Commission (NRC) memorandum entitled “Office Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and modified by Addendum dated January 18, 1979, provides key design criteria and regulatory guidance for new spent fuel storage racks.

NUREG-0800, SRP, Section 9.1.4, “Light Load Handling system,” also provides criteria for the staff to review the design and performance of the light load handling system used to move the spent fuel assembly into the shipping cask. Staff application of the criteria is focused on, among other things, the adequacy of the methods and equipment for transferring stored fuel to the spent fuel shipping cask. The NUREG, in Section 9.1.5, “Overhead Heavy Load Handling Systems,” also provides criteria for the staff to review the design and performance of the overhead heavy load handling systems used in moving all heavy loads, i.e. loads weighing more than one fuel assembly and its associated handling device. Staff application of the criteria in SRP Section 9.1.5 is in accordance with NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” Accordingly, the staff review is focused on (1) the performance and adequacy of the design, fabrication, installation, inspection, and testing of the load handling system; (2) the adequacy of the procedures for using the load handling system and performing load handling operations; and (3) the adequacy of the safe load paths and operator training.

## 2.2 Background

The spent fuel pool at DBNPS has a current storage capacity of 735 SFAs in 13 existing low-density rack modules (12 modules for storage of spent fuel and one module for storage of failed fuel). During the proposed full re-rack, DBNPS will replace all 13 existing low-density rack modules (735 SFAs) with 21 new high-density rack modules (1624 SFAs) designed and installed by the licensee's contractor, Holtec International.

DBNPS began Cycle 12 (May 1998) with insufficient storage capacity in the spent fuel pool to offload the entire reactor core (177 fuel assembles). Due to the need to perform its 10-year inservice inspection activities, DBNPS was required to conduct a full core offload during the 12th refueling outage (RFO) in April 2000. By letter dated May 21, 1999, the licensee proposed to revise the TS to support an increase in the spent fuel storage capacity from 735 to 1024 fuel assemblies using the spent fuel cask pit to temporarily store 289 SFAs until the spent fuel pool is fully re-racked. The NRC Safety Evaluation supporting license Amendment No. 237, dated February 29, 2000, approved this request and enabled DBNPS to fully offload the entire reactor core (177 fuel assemblies) and perform its 10-year inservice inspection activities during the spring 2000, 12th RFO. More specifically, Amendment No. 237 approved (1) the use of the cask pit to temporarily store up to 289 SFAs, and (2) the SFP cooling plan to re-rack the SFP at DBNPS during the 13th RFO in year 2002 to allow an increase in the SFP storage capacity from 735 SFAs to 1650 SFAs.

The approval of temporary storage of SFAs in the cask pit involved the use of four spent fuel storage rack modules (N1, N2, N3, and N4) containing a total of 289 storage cells. Two rack modules (N1 and N2), already installed, consisting of 153 of the 289 storage cells, were used for the full core offload during the 12th RFO and are currently being used to store SFAs. The remaining two racks consisting of 136 storage cells were to be installed during the full re-rack to support fuel shuffling.

In this submittal, DBNPS proposes to (1) replace all existing spent fuel storage racks in the DBNPS SFP; (2) use the cask pit to support the shuffling of fuel; (3) allow temporary storage of fuel in the transfer pit (for fuel shuffling) during SFP re-racking; and (4) relocate the cask pit and transfer pit racks to the SFP to complete the re-rack project.

## 2.3 Evaluation

### 2.3.1 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, surveillance requirement (SR) 4.9.7, and associated Bases 3/4.9.7 include provisions that prohibit movement of loads greater than 2430 pounds (heavy loads) over fuel assemblies in the SFP or in the cask pit. The proposed revisions to the LCO, SR, and associated Bases will prohibit the movement of heavy loads over fuel assemblies stored in the SFP, the cask pit, or the transfer pit.

The existing SR 4.9.7 is revised to require that except for a SFA, the licensee will verify that loads are not greater than 2430 pounds prior to being moved over fuel assemblies in the SFP, cask pit or transfer pit.

Furthermore, the TS and SR are revised to indicate that an impact cover weighing 2430 pounds may be moved over fuel assemblies in the cast pit (to install the cask pit cover) provided that it is administratively controlled. Other loads weighing more than 2430 pounds may be moved over the cask pit only when the impact cover is installed. However, administrative controls must be established to limit the loads to ≤ 17,350 pounds (the heaviest rack to be lifted including lift rig, rigging, and temporary hoist) and to limit the height that the load may travel over the impact cover to a height based on the design of the impact cover.

The proposed revisions to the LCO, SR and associated Bases are consistent with the design-basis fuel handling accident analysis in the Davis-Besse Updated Safety Analysis Report (USAR) Section 15.4.7 "Fuel Handling Accident," that was established to (1) limit any release to outside containment due to a dropped load that could damage 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 TS provision to allow an administratively controlled impact cover over the cask pit to preclude a potential load drop, enables the licensee to install the impact cover to provide protection of fuel assemblies stored in the cask pit during fuel handling operations. The TS provision to administratively limit loads up to 17,350 pounds to be moved at heights (to be determined by the licensee) over the impact cover, will enable the licensee to assure that the structural integrity of the impact cover to protect spent fuel assemblies stored in the cask pit in the event of a load drop is maintained.

The proposed weight limit of 2430 pounds for loads that can be moved over fuel assemblies stored in the transfer pit, is consistent with the existing TS which imposes the same weight limit on loads that can be moved over the SFP and the cask pit, therefore, it is acceptable. The provision to allow the impact cover to be installed over the cask pit enables the licensee to protect SFAs stored in the cask pit and is acceptable. The provision to administratively limit the weights of loads that may be moved over the impact cover is acceptable because it enables the licensee to reduce the potential for a dropped load to breach the structural integrity of the impact cover which is designed to protect SFAs stored in the cask pit. These provisions are consistent with the guidelines in NUREG-0612 and, therefore, are acceptable to the staff.

### 2.4 Heavy Load Handling Operations

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, the NUREG 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 and safety-related equipment.

As discussed above, TS restricting loads in excess of 2430 pounds from travel over SFAs in the SFP and the cask pit, will be extended to the transfer pit. Furthermore, the licensee is proposing, through use of administrative controls, to install an impact cover over the cask pit to protect the SFAs stored in the cask pit against potential drops of heavy loads. Also, the licensee is proposing to allow loads $\leq$ 17,350 pounds to be moved over the impact cover when the lift height of the load does not exceed the maximum height (to be determined by the licensee) wherein the structural integrity of the impact cover and the fuel assemblies stored in the cask pit is not jeopardized.

2.4.1 Hoisting System

The Spent Fuel Cask Crane (SFCC), the special lifting device, and non-customized lifting devices (i.e., slings), as used in re-rack of the cask pit, were previously approved in the staff's safety evaluation dated February 29, 2000, as being in accordance with the guidelines in NUREG-0612. However, in this complete re-rack of the SFP, the SFCC (140 tons) will be used in conjunction with a temporary crane (hoist) and special lifting devices for removal and installation of the new storage racks in the SFP, cask pit and transfer pit. Due to the limited travel of the SFCC, the temporary crane will be used to position existing racks for removal by the SFCC and to complete final positioning of new racks that are placed by the SFCC in the SFP. The temporary crane uses the existing rails of the fuel handling bridge crane, spanning the SFP, the cask pit and the transfer pit, and will be moved along the rails by the fuel handling bridge.

The licensee states that the temporary crane (hoist) is designed to the requirements of Crane Manufacturers Association of America (CMAA) -70 and the American Institute of Steel Construction (AISC) Manual. It also meets the intent of NUREG-0612 in that it satisfies the following:

- It lifts and moves the racks at several inches above the SFP floor, and is not used to lift any heavy loads out of the SFP nor over safety-related equipment;
- A rack drop from the temporary crane is bounded by the analyzed rack drop from 46 feet above the pool floor;
- Procedures govern operation of the temporary crane to ensure safe load paths;
- Trained personnel operate the crane; and
- The crane is manufactured as safety-related equipment and load tested to 120 percent of the maximum lifted load.

In addition, the crane was seismically analyzed for deformation and tipover, and the results indicated that gross failure of the crane will not occur, and tipover is not credible.

Based on the above, the staff concludes that the design, testing, and operation of the SFCC coupled with the hoist and lifting rig will enable the licensee to handle heavy loads with little to no risks to safety during the complete SFP re-rack operation. Therefore, the use of the lifting system during the re-rack operation is acceptable.

## 2.5 Fuel-Handling Accident (FHA) Analysis

USAR Section 15.4.7, "Fuel Handling Accident," evaluates the consequences of a FHA outside containment, including the SFP, cask pit, and transfer pit. This analysis assumes damage to 56 of 208 fuel rods with 72 hours decay-time and is bound by the FHA analysis inside containment which assumes that all 208 fuel rods with 72 hours decay-time are damaged. The licensee states that the SFP re-rack FHA analysis is bound by the FHA analysis inside containment, therefore, the design-basis FHA analysis will not change.

The elevation of the cask pit floor is 6'-6" lower than the elevation of the floor of the SFP and transfer pit. Therefore, although the drop of a SFA in the cask pit (approved in the safety evaluation dated February 29, 2000) bounds the drop of a SFA in the SFP and transfer pit, the licensee analyzed the drop of a SFA using the higher floor elevation to obtain more realistic results.

Section 7.0 of the Holtec Report, which was attached to the licensee's submittal, included two accident scenarios involving a shallow and a deep drop of a SFA into the SFP and transfer pit. The two accident scenarios were evaluated using a weight of 2,482 pounds dropped vertically from the peak lift height of 8.178 feet above the cask pit racks. The impact of the shallow drop resulted in a deformation of 8.75 inches into the fuel racks and < 50 percent blockage in the damaged SFA storage cells. The impact of the deep drop of a SFA from 8.178 feet onto the .75 inch thick baseplate of the rack (which is approximately 5.75 inches from the SFP liner) resulted in a maximum deformation of the baseplate of 3.36 inches. The SFP liner was not breached while the underlying concrete structure experienced some localized damage.

## 2.6 Load Handling Accident Analysis

Although all the load drop scenarios postulated to occur in the SFP and transfer pit are identical to the accidents postulated to occur in the cask pit (approved in the staff's safety evaluation dated February 29, 2000), the licensee re-analyzed the effect of the drop of a spent fuel storage rack. However, in this evaluation, the licensee analyzed a drop of the heaviest rack (14,030 pounds vs 12,150 pounds assumed for the cask pit analysis) from 46 feet above the floor of the SFP.

A deep drop of a rack (14,030 pounds) from 46 feet above the floor of the SFP and transfer pit resulted in displacement of the liner and some localized crushing of the SFP concrete slab, however, the liner was not pierced. Therefore, the licensee concluded that a dropped rack would not lead to catastrophic water leakage from the SFP and transfer pit.

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. As previously approved for the use of the cask pit, the licensee will implement measures using administrative controls and procedures to preclude load drop accidents in the SFP and transfer pit.

The above analyses are adequate to determine and characterize the nature and scope of potential damage to the SFAs, racks, SFP, and transfer pit floor following an accident involving a dropped SFA or rack during the complete SFP re-rack. The analyses indicated that neither the FHA nor load handling accidents will result in damage sufficient to cause a release of radioactive material beyond the limits set forth in 10 CFR Part 100. Furthermore, the potential damage from such accidents will not result in water leakage that could uncover the fuel in the SFP.

## 2.7 Thermal-Hydraulic Analysis

### 2.7.1 Spent Fuel Pool Cooling

The SFP cooling system (SFPCS) transfers decay heat from SFAs stored in the SFP to the component cooling water (CCW) system. The SFPCS consists of two half-capacity recirculating pumps, two half-capacity heat exchangers, associated piping, valves, instrumentation, and controls. The SFPCS heat exchangers are shell and tube units; the cold shell side flow is supplied from the CCW system and the hot tube side water is from the SFP. The SFPCS is designed to maintain the SFP water at or below 125 $^{o}$F with both cooling trains operating.

The decay heat removal system (DHRS) serves as the Class I backup cooling system to the SFPCS. The DHRS consist of two recirculating pumps, two heat exchangers, associated piping, valves, instrumentation, and controls. The DHRS heat exchangers are also shell and tube units and cooled by the CCW system. One DHRS train alone is capable of maintaining the SFP water temperature at or below 147 $^{o}$F with a decay heat load of $30.0 \times 10^6$ Btu/hr from an unplanned core discharge.

During partial-core discharge, one train of SFPCS, including one pump, one heat exchanger, and associated valves and piping, provide cooling for the SFP if a DHR train is available for SFP cooling. Both SFPCS trains are required to be functional if neither DHR trains are available for SFP cooling. During full-core discharge, both SFPCS trains and one DHR train is required to cool the SFP until the completion of fuel transfer. The DHR train may be used for other outage evolutions if there are provisions to ensure that it is readily available to support SFP cooling.

In the previously discussed submittal dated May 21, 1999, DBNPS presented the thermal-hydraulic analysis for the cask pit and the SFP in anticipation of the current re-rack of the SFP to increase the storage capacity from 735 to 1624 SFAs. As stated above, the February 29, 2000, safety evaluation, approved the DBNPS thermal hydraulic analysis for the future proposal to perform a complete re-rack of the SFP using the maximum density storage racks. In that safety evaluation, for the case of partial core discharge 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 $^{0}$F which is still below the SRP temperature limit of 140 $^{0}$F for SFP. For the case of abnormal full core discharge scenario with one train of DHR system operating, the calculated peak SFP temperature is 151.42 $^{0}$F which is below the SRP guidance for the SFP water temperature limit (pool boiling) during unplanned full core offload outages. That analysis was based on storing

1650 SFAs as a bounding SFP storage capacity value. Therefore, DBNPS' current proposal to increase the SFAs stored in the SFP to 1624 SFAs is bounded by the previously accepted thermal hydraulic analysis for the SFP and cask pit and is acceptable to the staff.

### 2.7.2 Cask Pit Cooling

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 a three foot thick concrete wall dividing the two areas.

The currently licensed storage capacity is 1024 SFAs (735 in the SFP and 289 in the cask pit area). As approved in Amendment No. 237 dated February 29, 2000, the cask pit consisting of 289 SFAs in four high density racks with a total heat output of 252,200 watts (861,006 Btu/hr), would be adequately cooled via natural circulation and flow through the cask pit gate. During the proposed complete re-rack of the SFP, the cask pit area will be used to support the shuffling of SFAs in the SFP, and therefore, it will contain fuel during most of the re-racking. However, the four racks in the cask pit will be relocated to the SFP. The heat loads resulting from re-racking the SFP will not affect the previous analysis. Therefore, the thermal-hydraulics of the cask pit and the SFP is bounding and the use of the cask pit during the re-racking operation is acceptable.

### 2.7.3 Transfer Pit Cooling

The licensee states that, if required, a single rack will be placed in the transfer pit for temporary storage of up to 90 SFAs during the re-rack. The transfer pit is normally flooded, located on the west side of the SFP, and separated from the SFP by a three foot thick wall. A three foot wide slot in the wall (which houses a gate) connects the transfer pit to the SFP. The gate can be installed in the slot and closed to allow for draining the pit without draining the SFP.

Although, the thermal analysis of the transfer pit with up to 90 SFAs is similar to and bound by that of the cask pit, the licensee performed an analysis of the transfer pit water temperatures. The gate is normally used to isolate the transfer pit from the SFP. However, during partial core discharge or full core offload operations or when SFAs are stored in the transfer pit, the gate will be open. In the analysis, the licensee assumes that the gate is both closed and open, and passive heat would be lost to the building environment from the water's surface.

Since the proposed increase in the SFP storage capacity would result in the increase of SFP heat load for all discharge scenarios that could impact the transfer pit, the evaluation of the water temperatures in the transfer pit addresses the six discharge scenarios that were previously evaluated for the bulk pool thermal-hydraulic evaluation of the SFP as follows:

Scenario 1: Planned Partial Core Discharge - involves a discharge of 72 SFAs at a rate of four SFAs per hour into the SFP, 150 hours after reactor shutdown, and following two years at full power with 1609 SFAs stored in the SFP prior to the discharge. Two SFPCS trains (two pumps and two heat exchangers) are aligned for cooling.

Scenario 2: Planned Partial Core Discharge - involves a discharge of 72 SFAs at a rate of four SFAs per hour into the SFP, 150 hours after reactor shutdown, and following two years at full power with 1609 SFAs stored in the SFP prior to the discharge. One SFPCS train (one pump and one heat exchanger) is aligned for cooling.

Scenario 3A: Unplanned Full Core Off-load Discharge - involves a discharge of 177 SFAs at a rate of four SFAs per hour into the SFP, 150 hours after reactor shutdown, and following 65 days at full power with 1537 SFAs stored in the SFP prior to the discharge. Two SFPCS trains (two pumps and two heat exchangers) are aligned for cooling.

Scenario 3B: Planned Full Core Off-load - involves a discharge of 177 SFAs at a rate of four SFAs per hour into the SFP, 150 hours after reactor shutdown, and following two years at full power with 1537 SFAs stored in the SFP. Two SFPCS trains (two pumps and two heat exchangers) are aligned for cooling.

Scenario 4A: Unplanned Full Core Off-load - involves 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 following 65 days at full power since the last discharge. A total of 1537 SFAs are stored in the SFP prior to the discharge. One train of the DHR system (one pump and one heat exchanger) is aligned for cooling.

Scenario 4B: Planned Full Core Off-load - Involves 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 following two years at full power since the last planned partial core discharge. One DHR system train (one pump and one heat exchanger) is used for SFP cooling.

#### 2.7.3.1 Transfer Pit Cooling - Gate Closed

As stated previously, the transfer pit is normally isolated from the SFP by a closed gate. However, the gate will be kept open during partial core discharge or full core offload operations or with SFAs stored in the transfer pit, and natural circulation of SFP water will be relied upon to maintain the water temperature limits for the transfer pit. The licensee stated that, on an as-needed basis, the transfer pit will only be used to support fuel shuffling during the re-rack operation and upon completion of the re-rack, the transfer pit will not house any SFAs nor fuel racks.

As stated by the licensee, in the analysis to evaluate the water temperature in the transfer pit with SFAs stored in the transfer pit and the gate closed, the total heat load in the transfer pit is limited to 88,110 watts (300,806 Btu/hr) in accordance with the requirements in the DBNPS Technical Requirements Manual (TRM). Based on this heat load limit, the licensee determined that the maximum bulk transfer pit temperature is limited to 140 °F (less than 150 °F as specified by the American Concrete Institute (ACI-349) for nuclear related concrete structures), and the transfer pit maximum fuel cladding temperature is 218.85 °F. Because this is less than the local boiling temperature of 239 °F at the top of the racks and the heat load limit of the TRM maintains the transfer pit bulk temperature at or below 140 °F, the licensee concluded that boiling in the transfer pit with the gate closed, will not occur.

In addition, the estimated evaporation rate with the heat load of 88,110 watts is .542 gpm. The licensee estimated that, at the rate of .542 gpm, it would take 5 days for the transfer pit normal operating water level to drop to the level where the TS action statement would be effective. The TS action statement requires that if the water level drops below 23 feet above the top of the fuel racks, then makeup water is to be provided. Makeup water can be provided to the transfer pit from the borated water storage tank (BWST) by gravity flow via the BWST pumps. Therefore, the means and time (five days) to restore transfer pit water is adequate. The staff agrees with the analysis and its results.

2.7.3.2 Transfer Pit Cooling - Gate Open

The licensee analyzed the impact of the SFAs stored in the transfer pit on transfer pit water temperatures when the gate is open and without forced circulation cooling. Based on its evaluation, the licensee stated that the temperatures between the transfer pit and the SFP will equalize within approximately 4 °F when the gate is removed and natural circulation between the regions occurs. The staff reviewed the two scenarios previously evaluated in license Amendment No. 237 dated February 29, 2000. For these two scenarios, the maximum temperatures in the transfer pit would be approximately 137 °F for the scenario of planned partial core discharge refueling outages and 155.5 °F for the scenario of unplanned full core off-load outages. These temperatures are well below the guidance of SRP Section 9.1.3 for SFP water temperature limits.

The analysis to evaluate the temperature in the transfer pit with the gate open similarly limits the total heat load in the transfer pit to 88,110 watts (300,806 Btu/hr) in accordance with the requirements of the DBNPS TRM. Accordingly, the maximum temperature in the transfer pit is limited to 140 °F and the total decay heat load of any SFAs temporarily stored in the transfer pit is not to exceed 88,110 watts. Based on this limit, the licensee determined that the transfer pit maximum fuel cladding temperature is 234.27 °F. Because of the heat load limit and the fact that this is less than the local boiling temperature of 239 °F at the top of the racks, the licensee concluded that boiling in the transfer pit with the gate open, will not occur. Based on the above discussion, along with the heat load limit controlled by the TRM, use of the transfer pit for the temporary storage of the SFAs during the DBNPS re-rack operation is acceptable.

2.8 Fuel Handling Area Ventilation

In the previously discussed staff safety evaluation dated February 29, 2000, the staff approved DBNPS analysis indicating that based on the most limiting full core discharge scenario the maximum calculated building temperature would be 103°F which is within the design capability of the fuel handling area ventilation system. Therefore, the staff concludes that proposed increase in SFA storage capacity will have little to no impact on the fuel handling area ventilation system.

### 2.9 Conclusion

Based upon the evaluation and results covering the areas regarding the control and handling of heavy loads, SFP thermal-hydraulics, and fuel handling area ventilation, as documented above, we conclude that the proposed revisions to DBNPS' Technical Specifications and the increase in SFP storage capacity are acceptable. Our review of the results and the methodology the licensee provided in the submittals find that the design and operation of the SFPCS meet the intent of the guidance described in the SRP for SFPs. Accordingly, the proposed revisions to the license will allow for the safe handling and continued safe storage of spent fuel in the SFP, during and following the re-rack operation.

## 3.0 CRITICALITY SAFETY EVALUATION

The proposed high density spent fuel storage racks in the Spent Fuel Pool (SFP) at the Davis-Besse Nuclear Power Station are designed to accommodate B&W 15x15 Mark B fuel assemblies. The proposed storage patterns for the fuel assemblies within the racks consist of three categories and the following restrictions:

1. Mixed Zone Three Region (MZTR) pattern, where fresh or low burnup assemblies (Category C assemblies) are separated from each other and from intermediate burnup fuel assemblies(Category B assemblies) by barrier fuel assemblies with high burnup (Category A assemblies).

2. Checkerboard (CB) pattern of empty cells, or cells with non-fuel bearing components, and cells with fresh or low burnup assemblies (Category C).

3. Homogeneous Loading (HL) pattern of intermediate burnup fuel assemblies (Category B).

The acceptable burnup and initial enrichment combinations for each of the three categories have been computed by the licensee and are included in the current submittal as part of the proposed technical specification changes. The supporting calculations and analyses demonstrate that the maximum effective multiplication factor, $k_{eff}$, including bias, and uncertainties shall be less than or equal to 0.95, with 95 percent probability at the 95 percent confidence level.

### 3.1 Analytical Methodology

The criticality analysis of the proposed high density storage racks submitted by the licensee is based on calculations performed with the Monte Carlo code MCNPa (Ref. 1). MCNPa is a continuous energy three-dimensional Monte Carlo code, which together with the ENDF/B-V cross-section library, is a standard and accepted tool for Spent Fuel Pool (SFP) criticality analysis (Ref. 2). An independent verification of the MCNPa calculations was performed with the three-dimensional multi-group Monte Carlo code KENO5a (Ref. 3) and a 238-group cross-section library based on ENDF/b-V (Ref. 4). Both codes have been benchmarked to 62 critical experiments and show excellent agreement at a 95 percent probability and a 95 percent confidence level.

The fuel composition at discharge was determined by depletion analyses during core operation with the two-dimensional multi-group transport theory code CASMO-4. This code is an acceptable depletion analysis code (Ref. 2) and has been extensively benchmarked against critical experiments, Monte Carlo calculations, reactor operations, and irradiated fuel compositions. To account for the effect of the axial burnup distribution on $k_{eff}$, the licensee developed a very conservative bounding axial power distribution for the analyses of the new racks. Further conservatism was introduced with regard to long-term changes in reactivity by setting the Xe-135 concentration to zero and by taking no credit for the continuous reactivity decrease in spent fuel due to the decay of Pu-241 and the growth of Am-242.

### 3.1.1 Normal Conditions

Criticality safety analyses, which take into account uncertainties due to burnup, tolerances, eccentric fuel positioning, water-gap spacing between racks, and racks and pool walls, were performed for the three different loading patterns (Mixed Zone Three Region, Checkerboard, and Homogeneous Loading) in the spent fuel pool. The moderator is un-borated water at a temperature of 20°C; the effect of the temperature reduction to 4°C (i.e. maximum reactivity) is included as an additional uncertainty in the calculations. The analyses result respectively in $k_{eff}$ values of 0.9470, 0.9324 and 0.9358 for the three loading patterns, and, therefore, are conservatively less than the regulatory limit on $k_{eff}$ of 0.95.

### 3.1.2 Accident Conditions

The analyses presented by the licensee of abnormal or accident conditions apply the double-contingency principle wherein credit for soluble boron may be assumed in the evaluation of other than loss of soluble boron accident conditions (Ref. 2). In this regard, administrative procedures, to assure the presence of soluble boron during fuel handling operations, preclude the possibility of the simultaneous occurrence of two independent accident conditions.

The largest reactivity increase, has been established to occur for a fresh fuel assembly inadvertently loaded into a cell with the remainder of the rack loaded in a checkerboard pattern. Calculations performed for this accident condition demonstrate that 630 ppm of soluble boron is adequate to assure that a $k_{eff}$ of 0.945 is not exceeded. Similarly the regulatory limit of $k_{eff}$ of 0.95 could possibly be exceeded if a fresh fuel assembly of the highest permissible enrichment were to be accidentally placed outside of a storage rack adjacent to other fuel assemblies; in particular, in a corner formed by three storage racks. Analyses show that a soluble boron concentration of 450 ppm is adequate to assure that $k_{eff}$ does not exceed 0.945.

For the case in which a fuel assembly is assumed to be dropped during fuel handling, dropping an assembly into an unoccupied cell could result in a localized deformation of the baseplate of the rack. A resultant deformation of the baseplate could potentially result in the active fuel height of that assembly not being completely covered by the Boral. Structural and neutronic analysis has shown that the excess reactivity introduced by such a localized deformation can be overridden by a soluble boron concentration of 55 ppm, and, thereby, assure that $k_{eff}$ of 0.945 is not exceeded. Administrative procedures provide that soluble boron remains no lower than

630 ppm.  Therefore, the computed $k_{eff}$ for the three racks meets the regulatory limits and it therefore, acceptable.

### 3.2 Conclusions

In view of the above considerations, the staff finds that the criticality analyses presented by the licensee to qualify the high density rack modules for storage of fuel assemblies in one of three different loading patterns:  Mixed Zone Three Region, Checkerboard, and Homogeneous Loading are acceptable.  The criticality analyses support the new TS given by TS Figure 3.9-3 which provides the category-specific burnup/enrichment limitations for the proposed loading patterns. The staff concludes that this specification is acceptable.  There is reasonable assurance that the new technical specification will result in facility operation within the regulatory limits of 10 CFR 50.68.

## 4.0 STRUCTURAL MATERIALS

### 4.1 Introduction

U.S. Tool and Die, Inc. (UST&D), Holtec International Designated manufacturer, will be the manufacturer of the new storage rack arrays proposed for use in the SFP.  Two of the racks have been previously fabricated and installed in the cask pit area adjacent to the spent fuel pool.  The NRC approved a license amendment issued on February 29, 2000, for the use of the cask pit area to store spent fuel racks.  The principal construction materials of these racks are American Society of Mechanical Engineers (ASME) SA-240-304 stainless steel and ASME SA-564-630 for the adjustable support spindles, including a Type 304 stainless steel sheathing that contains Boral ™ as the neutron absorber material.  The fabrication process will include skip and spot welding with ASME type 308 welding material.  These free-standing, self-supporting racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel (B&PV) Code.

### 4.2 Evaluation

#### 4.2.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel rack arrays are contained in Table 1 (below):

| Structural Component | Material |
|---|---|
| Cell assembly sheet metal | ASME SA-240-304 |
| Baseplate | |
| Cell connecting bar stock | |
| Internally threaded support pedestal | |
| Externally threaded spindle for support pedestal | ASME SA-564-630 |
| Weld Material | ASME Type 308 |

**Table 1 - Structural materials used in Holtec International spent fuel racks at Davis-Besse Unit 1.**

These materials have a history of in-pool usage. These materials are of proven durability and are compatible with the spent fuel assemblies and spent fuel pool water environment. Therefore, the staff finds the structural materials proposed for use in this application acceptable.

4.2.2 Neutron Absorber Material

Boral$^{TM}$ is the neutron absorbing material used in the new spent fuel pool rack arrays. Boral$^{TM}$ is a hot-rolled ceramic-metal (cermet) of aluminum and boron carbide clad in 1100 alloy aluminum. Boron carbide has a high boron content and is physically stable and chemically inert. Boral$^{TM}$ also provides a high cross-section for removing thermal neutrons. The 1100 alloy aluminum provides corrosion resistance through a hydrated aluminum oxide film that develops on the surface, within a few days, after exposure to the atmosphere or water. As this film forms, the corrosion layer penetrates the surface of the aluminum cladding only a few microns with no net loss of aluminum cladding. Hydrogen, a byproduct of the corrosion process, may cause deformation of the sheathing holding the Boral$^{TM}$ panels attached to the racks, resulting in deformation of the storage cells. To prevent this degradation from occurring, the Boral$^{TM}$ is contained in a sheathing cavity attached to the racks with spot welding, allowing the gases to vent. The neutron absorbing capability of Boral$^{TM}$ is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral$^{TM}$ in this application acceptable.

4.3 Conclusions

Based on the evaluation, the staff concludes that the materials used in the fabrication of the spent fuel rack arrays manufactured by UST&D are compatible with the spent fuel pool environment at Davis-Besse Unit 1. The degradation of the sheathing holding the Boral$^{TM}$ panels is prevented by venting the corrosion hydrogen byproduct. In addition, the corrosion process does not affect the neutron absorbing capability of Boral$^{TM}$. Therefore, the materials used in the new spent fuel rack arrays are acceptable.

## 5.0 STRUCTURAL MECHANICS

### 5.1 Introduction

The primary purpose of this review is to assure the structural integrity and functionality of the racks and the stored fuel assemblies subject to the effects of the postulated loads discussed in Appendix D of SRP Section 3.8.4, and fuel handling accidents.

### 5.2 Storage Racks

FENOC has proposed to increase the storage capacity from 735 fuel assemblies to 1624 fuel assemblies by replacing all of the existing racks with high density racks in the SFP structure. The storage racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee performed a 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 they are not attached to the floor or walls of the SFP structure. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the DYNARACK 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 was utilized to calculate nodal forces and displacements at the nodes, and to obtain 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 MR analyses, all racks were considered to be fully loaded with three different coefficients of friction ($\mu$=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the SFP floor. The 3-D MR analyses were performed to investigate the fluid-structure interaction effects between the racks and the SFP walls as well as those among the racks, and to identify the worst-case response for rack movement and for rack member stresses. For the 3-D SR analyses, the rack was considered to be fully loaded, half loaded, and almost empty with a coefficient of friction ($\mu$=0.8) between the rack pedestal and the SFP floor. The 3-D SR analyses were performed to investigate the stability of the rack with respect to overturning.

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 USAR. FENOC 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 (PSD) function compatible with the design response spectra as discussed in SRP Section 3.7.1.

A total of twelve (12) 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 0.43 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 SFP 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.

FENOC 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. FENOC demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code; thus, indicating that the weld connection design of the rack is adequate.

Based on: (1) FENOC's parametric evaluation (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 these stresses are compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, and (3) FENOC'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 loading conditions and, therefore, are acceptable.

### 5.3 Spent Fuel Storage Pool

FENOC analyzed the SFP to demonstrate the adequacy of the structure under fully-loaded fuel racks with all storage locations occupied by fuel assemblies. The fully-loaded structure was subjected to the load combinations specified in the DBNPS USAR.

The licensee's submittals indicate that the induced stresses due to the racks in the SFP are smaller than the ACI 349, Code Requirements for Nuclear Safety Related Concrete Structures, corresponding allowable stresses. In view of FENOC's stress calculations, the staff concludes that FENOC's structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading, and SSE loading conditions. Thus, the SFP design is acceptable.

### 5.4 Fuel Handling Accident

The following two refueling accident cases were evaluated by FENOC: (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 for the first accident case 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 for the second accident drop case show that damage will be restricted to a depth of 8.75 inches below the top of the rack, thus, indicating that the evaluation satisfies the acceptance criteria presented in the criticality safety evaluation. The staff reviewed the FENOC's analysis results and concurs with its findings. This evaluation is acceptable based on the FENOC's structural integrity conclusions supported by the parametric evaluations.

### 5.5 Conclusion

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

## 6.0 OCCUPATIONAL RADIATION PROTECTION AND RADIOACTIVE WASTE

## 6.1 Evaluation

### 6.1.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the installation of the replacement spent fuel rack modules in the DBNPS spent fuel pool with respect to occupational radiation exposure. For this modification, the licensee plans to remove all low density fuel racks from the spent fuel pool and add 21 high density fuel racks. A number of facilities have performed similar operations in the past. On the basis of lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed for approximately 4.1 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) principles. Workers performing the SFP re-racking operation will be given pre-job briefings to ensure that they are aware of their job responsibilities and precautions associated with the job. 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. Personnel radiation monitoring equipment, in addition to that routinely used (such as multi-badging and alarming and integrating devices), will be issued as required.

The licensee plans on using divers to assist in the removal of existing fuel racks, clearing obstructions, and installation of the new fuel rack modules in the SFPs. Prior to any diving operations, the licensee will conduct radiation surveys of the diving area. As part of these surveys, the licensee will verify the location of all spent fuel assemblies and irradiated objects located in the SFPs. Spent fuel assemblies and irradiated objects may be shuffled prior to divers entering the SFPs in order to ensure that doses to divers are maintained ALARA during the diving operations, but the licensee will not change the locations of the spent fuel assemblies or irradiated objects in the SFPs while the divers are in the water. The licensee will use visual and physical barriers to prevent divers from accessing spent fuel elements or other high radiation items or areas. Divers will be equipped with a calibrated electronic alarming dosimeter that will alarm underwater to warn the diver of high dose rates or when they exceed a predetermined dose limit. The licensee will also equip the diver with an underwater detector with a readout on the surface so that the licensee can monitor underwater dose rates in the diver work area and notify the diver of any significant dose rate changes. Continuous radiation protection coverage will be provided at all times while a dive is in progress and divers will have continuous voice communication with surface personnel providing dive support. The licensee will provide additional overhead and underwater lighting to support re-racking operations and provide a safe working environment for the divers. The licensee will use underwater cameras to monitor the movements of the divers.

The fuel integrity history at DBNPS indicates the possibility that discrete radioactive particles (DRPs) may be encountered during this fuel pool re-racking operation. The existing racks and equipment will be washed and monitored as they come out of the water in an effort to keep any DRPs in the fuel pool. The pool floor will also be surveyed upon removal of the racks. In addition, the diver’s remote readout underwater survey instruments and extremity monitoring should warn if a DRP is encountered.

The existing spent fuel pool filtration system will be used to maintain pool clarity. An installed skimmer will be used to recirculate and clean the SFP surface water. If needed, an auxiliary filter system or auxiliary skimmer may be utilized. The underwater vacuum system may be employed to remove extraneous debris, reduce general contamination levels prior to diving operations, and to assist in the restoration of pool clarity following any pressure washing operations.

The dose rates at the surface of the SFP are primarily a function of the concentration of radioactive materials suspended in the SFP water. This concentration, in turn, is determined by factors such as the pool volume, clean-up system flow rate, and recent movements of fuel or other components in the pool. During the SFP re-rack, activities in the pool are expected to marginally increase the dose rates above and around the SFP, cask pit and transfer pit. However, the licensee anticipates the radiologic conditions will be comparable to those seen during normal refueling operations. The long term impact of more fuel stored in the SFP will be minimal since radioactivity of the SFP water is only weakly dependent (if at all) on the total number of fuel modules stored in the pool.

The total number of spent fuel modules stored in the pool is a consideration for the radiation levels in areas adjacent to the SFP. Relative to the present fuel racks, the new racks will give a higher density of fuel next to the SFP walls. In addition, there is less clearance, and therefore, less water shielding, between the new racks and the SFP walls. The licensee has calculated the maximum expected dose rate of 12.2 mR/hr in any area room adjacent to the SFP. Since the minimum radiation zoning for these areas is less than, or equal to, 15 mR/hr, the SFP storage expansion will not require a change to any existing radiation zones.

In order to control the spread of contamination, the licensee will spray the low density spent fuel racks with water upon removal from the SFP. After the low density spent fuel racks have been allowed to drip dry, they will be sealed, loaded into special containers meeting all appropriate Department of Transportation shipping regulations, and transferred to another facility for processing and disposal. 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 conduct airborne radioactivity surveys and will utilize continuous air monitors in all normally occupied plant areas where the likelihood for airborne radioactivity exists during the removal and installation of the spent fuel rack modules in the SFPs.

On the basis of our review of the DBNPS license amendment, the staff concludes that the proposed increase in spent fuel storage capacity at DBNPS can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The licensee has estimated the that it will take approximately 4.1 person-rem of radiation dose (with approximately 2.1 rem to divers) to complete the project. The staff finds that this projected dose is in the range of doses for similar modifications at other plants and is therefore, acceptable.

### 6.1.2 Solid Radioactive Waste

Radioactive spent resins are generated by the processing of SFP water through the spent fuel pool purification system. No significant increase in the changeout frequency of demineralizer resin and filter media is expected due to the storage of additional spent fuel in the SFP. However, the licensee anticipates a one-time shortening of the resin change-out interval to support the re-racking operation. In addition, the low density spent fuel racks that will be removed from the SFP, and some miscellaneous items, such as portions of piping and other obstructions that will be removed to accommodate the new racks, will be processed and disposed of as radioactive waste. These low density spent fuel racks and other materials constitute a small percentage of the total solid radioactive waste generated at DBNPS. Hence, neither the SFP re-rack operation, nor the storage of additional spent fuel in the SFP, will result in a significant change in the generation of solid radioactive waste at DBNPS.

### 6.2 Conclusions

On the basis of our review of the DBNPS license amendment, the staff concludes that the proposed increase in spent fuel storage capacity at 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 will be minimized. The staff therefore finds the proposed increase in spent fuel storage capacity at DBNPS to be acceptable.

## 7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 8.0 ENVIRONMENTAL CONSIDERATION

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 October 11, 2001 (66 FR 51985), 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.

## 9.0 CONCLUSION

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.

Principal Contributors: Brian E. Thomas, DSSA/SPLB
April Smith, DE/EMCB
Roger Pedersen, DIPM/IOLB
Yuri Orechwa, DSSA/SRXB
Yong Kim, DE/EMEB

Date: October 19, 2001

REFERENCES

1. J. F. Briesmeister, Ed., “MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A,” LA-12625, Los Alamos National Laboratory (1993).

2. L. I. Kopp, “Guidance and the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” June 1998.

3. L. M. Petrie and N. F. Landers, “KENOVa - An Improved Monte Carlo Criticality Program with Supergrouping,” Volume 2, Section F11 from “SCALE: A Modular System for Performing Standardized Computer Analysis for Licensing Evaluation,” NUREG/CR-0200, Rev. 4, January 1990.

4. “SCALE 4.3: A Modular System for Performing Standardized Computer Analysis for Licensing Evaluation for Workstations and Personal Computers, Volume 0", CCC-545, ORNL-RSICC, Oak Ridge National Laboratory (1995).