Docket No. 50-334

November 1, 1993

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Mr. J. D. Sieber, Vice President
and Chief Nuclear Officer
Nuclear Power Division
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

Dear Mr. Sieber:

SUBJECT: ISSUANCE OF AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE DPR-66 IN RESPONSE TO CHANGE REQUEST NO. 202 REGARDING SPENT FUEL POOL SYSTEM (TAC NO. M84673)

The Commission has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1, in response to your application dated November 2, 1992, as supplemented in letters dated February 23, 1993, June 28, 1993, July 9, 1993, August 16, 1993 (two letters), September 3, 1993, September 8, 1993, and October 8, 1993.

This amendment revises Technical Specifications (TSs) 3/4.9.14, the Bases section for 3/4.9.14, and TS 5.6.1 and 5.6.3. The spent fuel pool (SFP) storage capacity is increased to 1627 locations and divided into three regions of specified enrichment and burnup. Table 3.9-2 is added to restrict the enrichment and burnup for the third region of the modified SFP. The Bases sections are revised to provide a description of the basis for the changes. The Bases section is also revised to clarify the boron concentration uncertainty for the SFP.

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

Sincerely,

Original signed by:
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 178 to DPR-66
2. Safety Evaluation

cc w/enclosures:
See next page

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

November 1, 1993

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Nuclear Power Division
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. J. D. Sieber
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated November 2, 1992, as supplemented in letters dated February 23, 1993, June 28, 1993, July 9, 1993, August 16, 1993 (two letters), September 3, 1993, September 8, 1993, and October 8, 1993, 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.

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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. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 178 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 1, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 178

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3/4 9-14	3/4 9-14
3/4 9-15	3/4 9-15
-----	3/4 9-15a
B 3/4 9-4	B 3/4 9-4
B 3/4 9-5	B 3/4 9-5
5-5	5-5
5-6	5-6

3/4.9.14 SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.14 Fuel is to be stored in the spent fuel storage pool with:

a. The boron concentration in the spent fuel pool maintained greater than or equal to 1050 ppm when moving fuel in the spent fuel pool; and

b. Fuel assembly storage in Region 1 restricted to fuel with an enrichment less than or equal to 5.0 w/o U-235; and

c. Fuel assembly storage in Region 2 restricted to fuel which has been qualified in accordance with Table 3.9-1; and

d. Fuel assembly storage in Region 3 restricted to fuel which has been qualified in accordance with Table 3.9-2.

APPLICABILITY: During storage of fuel in the spent fuel pool.

ACTION:

a. Suspend all actions involving movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in the incorrect Region until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

b. Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 1050 ppm, until such time as the boron concentration is increased to 1050 ppm or greater.

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

SURVEILLANCE REQUIREMENTS

4.9.14.1 Prior to placing fuel or moving fuel in the spent fuel pool, verify through fuel receipt records for new fuel, or by burnup analysis and comparison with Table 3.9-1 or Table 3.9-2 for spent fuel, that fuel assemblies to be placed into or moved in the spent fuel pool are within the above enrichment/burnup limits.

4.9.14.2 Verify the spent fuel pool boron concentration is ≥ 1050 ppm:

a. Within 8 hours prior to and at least once per 24 hours during movement of fuel in the spent fuel pool, and

b. At least once per 31 days.

TABLE 3.9-1

BEAVER VALLEY FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U235 ENRICHMENT FOR STORAGE IN REGION 2 SPENT FUEL RACKS

Initial U235 Enrichment	Assembly Discharge Burnup (MWD/MTU)
2.0	2585
2.5	9551
3.0	15784
3.5	21643
4.0	27260
4.5	33710
5.0	40000

NOTE: The data in the above table may be either interpreted linearly or may be calculated by the conservative equation below. This equation provides a linear fit to the design burnup limits.

Minimum burnup, MWD/MTU = 12100 * E% - 20500
Where E = Enrichment ($E \leq 5\%$)

TABLE 3.9-2

BEAVER VALLEY FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U-235 ENRICHMENT FOR STORAGE IN REGION 3 SPENT FUEL RACKS

Initial U-235 Enrichment	Assembly Discharge Burnup (MWD/MTU)
2.348	0
2.5	1,605
3.0	6,980
3.5	11,682
4.0	16,239
4.5	20,672
5.0	25,000

NOTE: The data in the above table may be either interpreted linearly or may be calculated by the conservative equation below. This equation provides a best fit to the analysis burnup limits.

Minimum burnup, MWD/MTU $= -480 * (E\%)^2 + 12{,}900 * E\% - 27{,}400$
Where E = Enrichment ($E \leq 5\%$)

3/4.9.12 AND 3/4.9.13 FUEL BUILDING VENTILATION SYSTEM (Continued)

fuel building. The SLCRS flow is normally exhausted to the atmosphere without filtering, however, the flow is diverted through the main filter banks by manual actuation or on a high radiation signal.

3/4.9.14 FUEL STORAGE - SPENT FUEL STORAGE POOL

The requirements for fuel storage in the spent fuel pool ensure that: (1) the spent fuel pool will remain subcritical during fuel storage; and (2) a uniform boron concentration is maintained in the water volume in the spent fuel pool to provide negative reactivity for postulated accident conditions under the guidelines of ANSI 16.1-1975. The value of 0.95 or less for K_{eff} which includes all uncertainties at the 95/95 probability/confidence level is the acceptance criteria for fuel storage in the spent fuel pool.

The Action Statement applicable to fuel storage in the spent fuel pool ensures that: (1) the spent fuel pool is protected from distortion in the fuel storage pattern that could result in a critical array during the movement of fuel; and (2) the boron concentration is maintained at ≥ 1050 ppm (this includes a 50 ppm conservative allowance for uncertainties and 600 ppm for margin) during all actions involving movement of fuel in the spent fuel pool.

The Surveillance Requirements applicable to fuel storage in the spent fuel pool ensure that: (1) the fuel assemblies satisfy the analyzed U-235 enrichment limits or an analysis has been performed and it was determined that K_{eff} is ≤ 0.95; and (2) the boron concentration meets the 1050 ppm limit.

The reracked spent fuel pool consists of discrete Regions 1 and 2 with Region 2 further subdivided and identified as Regions 2 and 3. Region 1 is configured to store fuel with a nominal region average enrichment of 5.0 weight percent (w/o) with individual fuel assembly tolerance of + or - 0.05 w/o U-235. The most reactive of the Westinghouse 17 X 17 STD/Vantage 5H and OFA fuel assemblies yielded a maximum K_{eff} of 0.940 including all biases and uncertainties.

Region 2 racks are designed to store fuel with burnup consistent with its initial enrichment. A table of enrichment and corresponding required burnup is provided in the technical specification. A conservative value of the required burnup is given by the following linear equation:

> Minimum burnup for unrestricted storage in Region 2 in MWD/MTU = 12100 * E% - 20500, where E is the initial enrichment in w/o.

3/4.9.14 FUEL STORAGE - SPENT FUEL STORAGE POOL (Continued)

Storage cells in Region 2, which face the pool wall, are in an area of high neutron leakage and are capable of maintaining the K_{eff} below 0.95 with fuel that does not meet the foregoing burnup restriction. A separate calculation to establish the admissibility of storing low burnup fuel in these cells, designated Region 3, has been performed and a table of enrichment and corresponding required burnup is provided in the technical specification. This calculation was performed using the same analytical models and computer codes which were used in the high density rack design. A conservative value of the required burnup is given by the following linear equation:

> Minimum burnup for fuel storage in Region 3 in MWD/MTU = $- 480 * (E\%)^2 + 12{,}900 * E\% - 27{,}400$, where E is the initial enrichment in weight percent.

The maximum reactivity in Region 2 is 0.945 and in Region 3 is 0.946 if all cells are loaded with fuel with minimum allowable burnup. This includes all biases and uncertainties and appropriate allowance for uncertainty in depletion calculations.

3/4.9.15 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

The OPERABILITY of the control room emergency habitability system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The ambient air temperature is controlled to prevent exceeding the allowable equipment qualification temperature for the equipment and instrumentation in the control room. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the whole body radiation exposure to personnel occupying the control room to 5 rem or less, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9370 cubic feet at a nominal Tavg of 525°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed in a three region configuration. Region 1 racks are of the poisoned flux-trap type with the storage cells arranged at 10.82 inch pitch. The Region 2 and Region 3 racks are of the poisoned non-flux trap construction with a cell-to-cell pitch of 9.02 inches. The fuel will be stored in accordance with the provisions described in UFSAR Sections 3.3 and 9.12 to ensure a keff equivalent to ≤0.95 with the storage pool filled with unborated water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 750′ - 10".

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1627 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in Appendix "B" of the FSAR shall be designed and maintained to the original design provisions with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower shall be located as shown on Figure 5.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY
OHIO EDISON COMPANY
PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated November 2, 1992, Duquesne Light Company (DLC) requested changes to the Beaver Valley Unit 1 (BV-1) Technical Specifications (TS) to reflect installation of new spent fuel storage racks. The new racks would accommodate an increase in spent fuel assemblies beyond the existing storage capacity of the spent fuel pool (SFP). Additional information was provided to clarify the basis for the proposed change in letters dated February 23, 1993, July 9, 1993, August 16, 1993 (two letters), September 3, 1993, September 8, 1993, and October 8, 1993. In addition, in a letter dated June 28, 1993, the request was modified to permit three regions of storage instead of two in the SFP. The letters of clarification did not make a substantive change to the original amendment proposal, and it was not necessary to modify the No Significant Hazards Evaluation (NSHE) which was published in the Federal Register on February 4, 1993, (58 FR 7161). However, the June 28, 1993, supplement revised the proposed design and TS, and a revised NSHE was published on September 8, 1993 (58 FR 50977).

The present BV-1 SFP storage racks have a total storage capacity of 833 cells. These racks provide adequate capacity for storage of spent fuel while maintaining a full core reserve discharge capacity through 1996. In order to ensure that sufficient spent fuel storage capacity continues to exist at BV-1 beyond 1996, DLC plans to install high density storage racks which incorporate Boral as a neutron absorber in the cell walls, thereby allowing for more dense storage of spent fuel. The new racks would provide an ultimate storage capacity of 1627 locations (including two storage cans for defective fuel) and extend capability for full core discharge through the year 2013.

The following evaluation assesses the proposed change with regard to materials selection, criticality aspects, movement of heavy loads, thermal/hydraulic aspects, mechanical and seismic loading, and radiation protection.

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2.0 EVALUATION

2.1 Materials Selection

Structural, poison, and welding materials are evaluated in sections 2.1.1 through 2.1.3.

2.1.1 Structural and Welding Materials

The licensee has hired an outside contractor to perform a safety analysis for the proposed license amendment (No. 202). The contractor has selected the following structural and welding materials for use in the proposed storage rack modification:

- American Society of Mechanical Engineers (ASME) SA240-304 for all sheet metal stock - this is a type 304 stainless steel which conforms to ASME standard SA240

- ASME SA240-304 for the internally threaded support legs

- ASME SA564-630 for the externally threaded support spindle - this is a precipitation hardened stainless steel, which has been heat treated to 1100 °F

- Weld material - type R308L stainless steel conforming to ASME specification SFA 5.9

ASME SA240-304 stainless steel is a common austenitic alloy which has been used in nuclear applications. ASME Specification SA-240 requires that Type 304 stainless steel have a minimum yield strength of 30 ksi, and a minimum ultimate tensile strength of 75 ksi. The choice of type 304 stainless steel for the sheet metal stock and the internally threaded support legs is reasonable. The high nickel content stabilizes the austenitic phase of the steel at room temperature. The high chromium content imparts reasonable corrosion resistance to oxidizing effects of most electrolytes when at low concentration levels. The steel is, however, susceptible to corrosion in acidic solutions (pH < 7.0) containing chloride or fluoride anions. These anions can lead to pitting or stress corrosion cracking of the material. The corrosion effects by chloride or fluoride anions is not as pronounced in basic media (pH > 7.0).

The licensee has opted to use a SA564-630 steel, heat treated at 1100 °F, for the externally threaded support spindle. Type 630 (also called 17-4 PH steel) is a martensitic, precipitation hardened (PH) stainless steel. Martensitic, PH, stainless steels have increased strength, without suffering considerable loss of ductility. The corrosion resistance, however, is not quite as good as that of austenitic stainless steels. Heat treatments and aging of martensitic, PH stainless steels above 1025 °F increases their resistance to stress corrosion cracking.

The procured specimens have been manufactured to conform to ASME Section II specification, SA-564. SA-564 requires that a 630 steel heat treated at 1100 °F have a minimum yield strength of 115 ksi, a minimum tensile strength of 140 ksi, a minimum elongation (2 inch specimen) of 14%, and a minimum charpy impact energy of 25 ft-lb. This steel has been selected primarily for its high strength. This steel will also reduce the chance of galling with the Type 304 internally threaded support legs.

It should be noted that control of water impurities in nuclear plant SFP water is typically provided by the SPF demineralizers in the spent fuel cooling system. The demineralizers function to keep the chemistry of the SFP water approximately the same as that of the reactor coolant system, in order to minimize the probability of abnormal chemistry incursions during refueling operations when the two systems link together. Control of SFP chemistry, however, also serves to reduce corrosion effects by keeping the concentrations of water impurities at low levels. Therefore, stress corrosion cracking or pitting, induced by residual chloride or fluoride ions in the fuel pool, should not be a problem with either the SA240-304 stainless or SA564-630 precipitation hardened stainless steels.

The type R308L weld filler metal has been selected since it has a similar composition to the type 304 structural material. This material conforms to ASME Code Section II, Specification SFA-5.9, and is suitable in gas metal arc welding and gas tungsten arc welding applications of both the SA240-304 and SA564-630 steels.

2.1.2 Poison Material

- Boral - patented material produced by AAR Brooks and Perkins

Boral is a cermet composite material made of Type 1100 aluminum and boron elements. The composite panel consists of two outer type 1100 aluminum sheets which clad a sintered plate of boron carbide in a Type 1100 aluminum matrix. The type 1100 aluminum has sufficient strength (i.e., a 18 ksi ultimate tensile strength for the as rolled material) to justify its use in non-structural applications. The Type 1100 aluminum material imparts sufficient pitting and general corrosion resistance by forming an aluminum oxide layer on its surface when exposed to oxidizing environments. This oxide layer is stable in environments with a pH range of 4.5 - 8.5. The boron carbide particles are manufactured in accordance with ASTM Specification C-750 for nuclear grade materials. The boron carbide particles have been shown to have good structural compatibility with the Type 1100 aluminum matrix material.

Boral is typically manufactured in 0.177 inch and 0.265 inch thicknesses in accordance with AAR Brooks and Perkins Specification BPS-9000-01, "Item Specification for Boral, a Neutron Shielding Material." The manufacture of Boral panels of other thicknesses is done in accordance with the guidelines of

BPS-9000-02. BPS-9000-01 and -9000-02 fall within the confines of the manufacturer's quality assurance program for nuclear grade materials. The licensee intends to install the Boral sheets by freely inserting them between the 304 stainless steel walls of the rack assemblies and the 304 stainless steel sheaths which are to be welded to the wall. The licensee has stated in its safety evaluation that Boral sheets for the Region 1 and Region 2 (and 3) cells are being procured by special order according to the following dimensions, parameters, and tolerances:

Boral Parameter	Region 1 Cells	Region 2 (and 3) Cells
Boral Sheet Width	7.25 +- 0.13 inches	7.25 +- 0.13 inches
Boral Sheet Length	144 +- 0. inches	144 +- 0. inches
Boral Sheet Thickness	0.084 +- 0.004 inches	0.098 +- 0.004 inches
Boral B^{10} Areal Density	0.0248 g/cm^2 (0.023 g/cm^2 min.)	0.0320 g/cm^2 (0.028 g/cm^2 min.)

These parameters conform to those stated in the manufacturer's specifications and material information documents. The design of the sheath areas is such that the Boral Sheets and the surrounding sheath area walls will be 0.005 inches to 0.010 inches offset from each other. It is evident, however, that the insertion of the Boral panels will create a tight fit. A number of independent studies by industry organizations and by NRC contractors have shown that the reaction of Boral with water or moisture may generate hydrogen gas. This is evident from the following reaction between water and aluminum (Type 1100 in this case):

$$2Al(s) + (3+x)H_2O(l) \equiv Al_2O_3.xH_2O(s) + 3H_2(g),$$

which yields two products, a hydrated form of aluminum oxide and hydrogen. Production of hydrogen may result in deformation of the rack cells by imparting additional stresses on the walls. Information Notice 83-29, "Fuel Binding Caused by Fuel Bundle Deformation," was issued to alert the industry to this concern. The licensee has indicated in its submittal that holes at the corners of the sheath areas will create a sufficient vent path for any potential hydrogen which may be produced by a water-aluminum reaction.

The licensee has also created a Boral Surveillance Program to characterize the performance of the Boral panels during the remaining lifetime of the plant. This program is in accordance with the NRC Letter of April 14, 1978, to all nuclear power licensees, which stated that "Methods for verification of long-term material stability and mechanical integrity of special poison materials utilized for neutron absorption should include actual tests."

The licensee's Boral Surveillance Program calls for placing 10 Boral test coupons, suspended on a mounting jacket, in a designated cell that is surrounded by four spent fuel assemblies. The coupons will be taken from the same lot as that used for the manufacture of Boral in the construction of the racks. The mounting jacket will be constructed from the same Type 304 stainless steel alloy as is used in the construction of the fuel pool racks. Two additional, unjacketed coupons will be preserved as control specimens.

The surveillance program calls for removing and testing one coupon from the mounting jacket at the following intervals relative to the installation of the racks: 1st cycle, 2nd cycle, 4th cycle, 7th cycle, 10th cycle, 20th cycle, 30th cycle, and 40th cycle. During each of the first six cycles, the spent fuel assemblies surrounding the coupon mounting jacket will be replaced with freshly discharged spent fuel assemblies. This is done to ensure that the radiation exposure of the Boral coupons is slightly higher than that experienced by the Boral panels in the racks. From the seventh refueling cycle on, the surrounding spent fuel assemblies will remain in place. Each coupon, upon its removal from the mounting jacket, will be analyzed according to the following tests:

- Visual Observation and Photography
- Neutron Attenuation
- Dimensional Measurements (length, width, and thickness)
- Weight and Specific Gravity

The neutron attenuation and the dimensional measurements are the most important tests of the group since they are used to determine whether or not the coupons are exhibiting any signs of boron loss or structural deformation, respectively. The gravimetric analyses will be performed to augment the results of the neutron attenuation studies if boron loss is indicated.

2.1.3 Summary

DLC's license amendment request submittal indicates that material selection for the Beaver Valley spent fuel rerack modification has been satisfactorily thought out. With the exception of the internally threaded support spindle, the majority of the rack is to be constructed from a Type 304 stainless steel fabricated according to an approved ASME Section II specification. The internally threaded support spindle will be constructed from an ASME Section II approved precipitation hardened stainless steel. Boral is an acceptable poison material; however, since the Boral may generate hydrogen when in contact with water or moisture, care must be taken to provide a sufficient path to allow potential hydrogen generation to vent from the sheath area. Based on a review of the licensee's submittal, the staff has determined that the licensee's selection of structural, welding and poison materials meets current industry and regulatory standards and that these materials are acceptable for construction of the new rack modules.

2.2 Criticality Aspects

Three separate storage regions are provided in the SPF with independent criteria defining the highest potential reactivity in each of the three regions. Region 1 is designed to accommodate new fuel with a maximum enrichment of 5.0 weight percent (w/o) U-235 or spent fuel regardless of its discharge burnup. Region 2 is designed to accommodate fuel of various initial enrichments up to 5.0 w/o U-235 which have accumulated minimum irradiation levels within the acceptable burnup domain depicted in the upper curve of Figure 4.2.1. In addition, peripheral cells in Region 2 which face the pool wall are designated as Region 3 and designed to accommodate low burnup fuel of various initial enrichments up to 5.0 w/o U-235 which have accumulated minimum irradiation levels within the acceptable burnup domain depicted in the lower curve of Figure 4.2.1.

The analysis of the reactivity effects of fuel storage in Region 1, 2, and 3 was performed with the two-dimensional transport theory code, CASMO-3. Independent verification calculations were made with the KENO-5a Monte Carlo computer code using the 27-group SCALE cross-section library. Since the KENO-5a code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the BV-1 spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber thickness. These two independent methods of analysis (KENO-5a and CASMO-3) showed good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a calculations, a minimum of 500,000 neutron histories in 1,000 generations of 500 neutrons each were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-5a reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the BV-1 storage racks with a high degree of confidence.

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

(1) Unborated pool water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures.

(2) Assumption of infinite array of storage cells in all directions (except for the assessment of peripheral effects and certain abnormal conditions where neutron leakage is inherent).

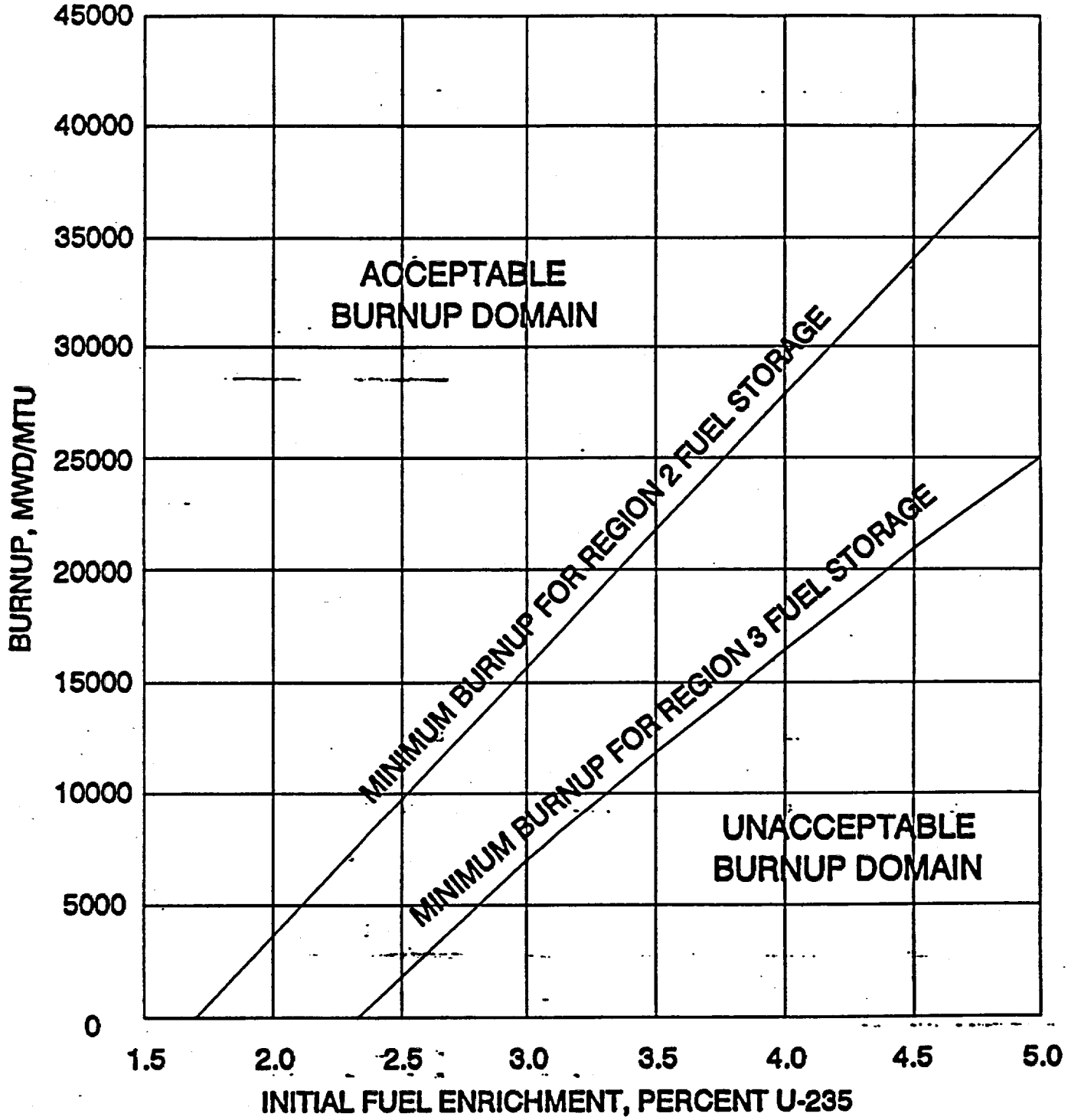

FIGURE 4.2.1 ACCEPTABLE BURNUP DOMAIN FOR REGIONS 2 and 3

(3) Neutron absorption effect of structural material is neglected.

In addition, the design basis fuel assembly was a Westinghouse 17x17 array of fuel rods containing UO_2 at a maximum initial enrichment of 5.0 w/o U-235. Since BV-1 contains Westinghouse Standard, Vantage-5H and optimized fuel assembly (OFA) fuel designs, calculations were made for each of these fuel types. The OFA fuel was determined to exhibit the highest reactivity in Region 1 and Vantage-5H, the highest in Region 2.

The staff has determined that appropriately conservative assumptions were made.

For the nominal storage cell design, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, and fuel enrichment and density tolerances were accounted for. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (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. The final maximum calculated reactivity resulted in a k-eff of 0.940 (Region 1), 0.945 (Region 2), and 0.946 (Region 3) when combined with all known uncertainties. This meets the staff's criterion of k-eff no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

The calculated maximum reactivity in Region 2 includes a burnup-dependent allowance for uncertainty in depletion calculations and, as mentioned above, provides an additional margin below the limiting k-eff criterion of no greater than 0.95. Although not included in the criticality analyses, subsequent decay of Pu-241 with long-term storage results in a significant decrease in reactivity. This will provide an increasing subcriticality margin and further compensate for any uncertainty in the depletion calculations. The attached Figure 4.2.1 shows that Region 2 can safely accommodate fuel of various initial enrichments up to 5.0 w/o and discharge burnups provided the combination falls within the acceptable domain illustrated by the upper solid line. This acceptable burnup domain is presented in tabular form in TS Table 3.9-1. This reactivity equivalencing method is the standard one used for storage rack reactivity evaluations and is acceptable.

Because of the high neutron leakage from the boundary cells which face the pool walls, additional calculations were performed to determine the burnup required for safe storage in these cells, designated Region 3. KENO-5a calculations for these cells, assuming fuel of 5.0 w/o initial enrichment irradiated to 25,000 MWD/MTU with the remainder of the racks filled with fuel of 40,000 MWD/MTU burnup, resulted in a reactivity less than 0.95. Therefore, these boundary cells can safely accommodate fuel of the enrichment-burnup

combination shown by the lower curve in Figure 4.2.1, and given in tabular form in TS Table 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 Figure 4.2.1 or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of approximately 1050 ppm of boron in the pool water required by the plant TS whenever fuel is stored in the SFP since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). 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 400 ppm boron would be adequate to assure that the limiting k-eff of 0.95 is not exceeded.

The following TS changes have been proposed as a result of the requested SFP reracking. Based on the above evaluation, the staff finds these changes acceptable as well as the associated Bases changes.

(1) TS 3.9.14.b has been modified to reflect the increase in allowable maximum enrichment to 5.0 w/o U-235 for unrestricted storage in Region 1.

(2) TS 3.9.14.d has been added to reflect allowable storage in the peripheral cells (Region 3) according to Table 3.9-2.

(3) TS Table 3.9-1 has been modified to reflect allowable storage in Region 2 for assemblies meeting the minimum burnup versus initial enrichment limits given. The tabular data may be either interpolated linearly or may be calculated by the limiting burnup equation shown below the Table.

(4) TS Table 3.9-2 has been added to reflect allowable storage in Region 3 for assemblies meeting the minimum burnup versus initial enrichment limits given. The tabular data may be either interpolated linearly or may be calculated by the limiting burnup equation shown below the Table.

(5) TS 5.6.1 has been modified to describe the new three-region SFP configuration. The Region 1 racks are of the poisoned flux-trap type with the storage cells arranged at 10.22 inch pitch. The Region 2 and 3 racks are of the poisoned non-flux-trap construction with a cell-to-cell pitch of 9.02 inches.

(6) TS 5.6.3 has been modified to reflect the increased fuel pool storage capacity to 1627 fuel assemblies.

Based on the review described above, the staff finds the criticality aspects of the proposed modifications to the BV-1 SFP storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

2.3 Movement of Heavy Loads

The present storage racks, 17 in number, are designed to contain fuel assemblies in 7 x 7 arrays. The new racks, 13 in number, have capacities ranging from one with a 7 x 11 capacity to one with a capacity of 11 x 15 spent fuel assemblies, with varied capacities between these extremes. Similarly, the weights of the new racks vary from 13,300 to 24,300 lbs. The new racks will be free standing. The existing racks are bolted to the SFP floor and braced to the walls. All of the present racks will be removed to make room for the new racks.

The licensee needs to store some fuel in the cask pit to accomplish the reracking process. The cask pit will be provided with a cover upon which the licensee will store empty racks, temporarily. The licensee plans to use a temporary crane to move racks into and out of the SFP, with temporary placement on the cask pit cover. The cask crane will remove old racks from the cover and place new racks on the cover for introduction into the SFP by the temporary crane during the reracking process.

2.3.1 Temporary Crane

The temporary crane will be assembled on site and will share the same rails as the present movable platform. It will be able to traverse the entire length of the SFP and the cask pit. The licensee provided details of its construction, showing that the crane had been designed in accordance with ANSI Standard B30.2.0-1976, "Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)."

The temporary crane has been designed for a nominal load of 150 tons before the stress for any component reaches the yield point. The ratio of the load capability of the crane to that of the load, approximately 15 tons, is 10/1. The licensee stated that the crane is capable of carrying a load considerably in excess of 150 tons before the crane would fail. The temporary crane will be physically connected to the fuel bridge crane for lateral stability. The fuel bridge crane also provides motive power to the temporary crane. While fuel shuffling is carried out, the temporary crane will be parked over the cask pit.

2.3.2 Cask Crane

The cask crane was derated from its original design load of 125 tons to 60 tons because the reeving on the crane was changed in order to increase the hook speed. The factor of safety of 5 still applies, giving it an ultimate capability of 300 tons (600,000 lbs.) before failure. This provides a factor of safety in excess of 20 when moving the heaviest racks with the special

lifting device (see section 2.1.3 below for discussion of special lifting device.). This crane cannot traverse any portion of the SFP.

2.3.3 Special Lifting Device

The licensee plans to use a special lifting device to move racks into and out of the SFP. This device, called a remotely engageable lift rig, has four independent legs, each of which engages into a square frame at the bottom; the frame is not designed to carry any load but merely to position the four legs in their proper locations.

The rig is designed in accordance with the requirements of ANSI Standard N14.6-1978, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg.) or More." The rig is designed to lift a load of 136 tons before the stress for any component reaches its ultimate value. When carrying the heaviest rack, weighing 13 tons, the ultimate factor of safety is 10.5.

In the event of failure of one of the legs, the ratio of the ultimate load capability of the three remaining legs to the maximum load would be 7.8. A leg failure should not result in undue load movement since the center of gravity (CG) of the lift point is to be aligned with the CG of the load before starting to move the load.

2.3.4 Lifting Devices Not Specially Designed

The licensee stated that such devices would be designed in accordance with the requirements of ANSI Standard B30.9-1971, "Slings." While the licensee did not provide detailed design characteristics, the licensee did state that "a hoist system with an ultimate load rating of at least 150 tons will be used to effect the load movements." Therefore, slings, with ultimate load ratings of at least 150 tons, would be used to move racks if necessary, thus providing a minimum safety factor of 10 before the slings (or any portion of the heavy load lifting system) would fail.

2.3.5 Cask Pit Cover

The licensee plans to store some spent fuel in the cask pit in order to expedite the reracking process. A cover will be installed over the pit and over fuel temporarily stored there. The cover has a two-fold purpose: to protect the fuel in the pit and to serve as a place where racks can be stored, temporarily, during moves into and out of the SFP. Racks will be moved out of the pool by the temporary crane and placed on the pit cover as a temporary expedient. The cask pit crane will be used to move the racks from the cask pit cover to other locations away from the SFP.

The licensee stated that the cover could withstand a drop of the heaviest rack from a level three (3) inches above the cover. In order to assure the three inch limitation, the licensee will provide procedures specifying this limitation.

2.3.6 Safe Load Paths

The licensee stated that the temporary crane, when moving any heavy load into or out of the SPF, will maintain a minimum lateral clearance of at least 3 feet from any active fuel. Stops will be installed on the fuel handling crane tracks and on the temporary crane hoist to attain such separation.

Safe load paths will also be used by the cask crane when moving racks from and to the cask pit cover.

2.3.7 Other Heavy Load Handling Features

The licensee will provide other features in order to minimize the possibility of damage as a result of a load handling accident. The licensee plans the following:

(1) To avoid operator errors, comprehensive training will be provided to the installation crew using videotapes of the actual lift rigs built for the project.

(2) To minimize rigging failures, the lifting rig has four independent load paths, with the ability to retain the load in the event of failure of one leg.

(3) The rack designer will develop a set of rack inspection points to eliminate rework or erroneous rack installation.

(4) The licensee plans to develop over twenty operating procedures designed to cover all aspects of the rerack operation.

(5) The cranes and hoists used in the project will be given a preventive maintenance checkup and inspection within a period of 3 months prior to the start of the reracking process.

(6) The licensee plans to fully inspect the fuel handling crane and to relubricate its moving parts prior to the start of reracking.

2.3.8 Summary

The temporary crane is capable of lifting 150 tons, as much as 10 times the maximum load to be carried (the heaviest rack and lifting device is about 15 tons). The temporary crane will be attached to the fuel handling crane for motive power and for protection against lateral movement. This attachment, together with wheel restraints, will prevent the temporary crane from leaving the rails and falling into the SFP in the event of an earthquake.

The cask crane is capable of lifting loads as great as 20 times (300 tons) the weight of the heaviest rack and special lifting device before failure.

The cask pit will be used for temporary storage of spent fuel assemblies. A cask pit cover will provide for temporary storage of racks leaving and entering the SFP; it will be used as a point for transferring rack movement from the temporary to the cask crane and vice versa, depending upon whether old racks are leaving or new racks are entering the SFP.

The licensee will maintain a minimum 3-foot lateral separation between spent fuel and all empty racks being moved when the latter is above the former. Safe load paths will be employed by the cask crane when moving.

The licensee will train installation crews, will have the rack designer provide inspection points for the new racks to eliminate rework and aid in installation, and will develop operating procedures to cover all reracking operations. The fuel handling crane will be inspected and lubricated before starting.

The staff considers that heavy loads concerns have been addressed in an acceptable manner.

2.4 Thermal/Hydraulic Aspects

2.4.1 Spent Fuel Pool Cooling and Purification System

This system consists of two parts; the cooling system, and the purification system part. The cooling system has two trains. Each train has a pump with a flow capacity of 750 gpm and a heat exchanger with a nominal design duty of 11.4 million Btu/hr (11.4 MBtu/hr). The system can be arranged to use either pump with either heat exchanger.

The SFP purification system portion is comprised of two 400 gpm pumps, two 400 gpm pool filters, a skimming assembly, and a mixed bed deionizer. This system portion may be so arranged as to use either filter with or without the deionizer. The deionizer design temperature is 200 °F.

2.4.2 Cases Analyzed

The licensee analyzed four cases in determining the maximum SFP bulk coolant temperatures. These are:

(1a) Normal refueling load of spent fuel assemblies discharged, one train of SFP cooling system operating

(1b) Same as 1a with both SFP cooling trains operating

(2) Full core (157 spent fuel assemblies) discharged, both SFP cooling trains operating

(3) Refueling load (72 fuel assemblies) discharged, then full core offload approximately 60 days later, both SFP cooling trains operating

2.4.3 Decay Heat Generation

The licensee stated that the 22 batches constituting 1524 fuel assemblies, which would be in the reracked pool before losing the capability to unload the full core, would generate heat equivalent to 7.65 million Btu/hr. The total pool heat load calculated by the licensee for the four cases at the time the SFP coolant attained its maximum bulk temperature is shown below (Table 1):

TABLE 1

SFP DECAY HEAT AND MAXIMUM BULK COOLANT TEMPERATURE

CASE	MAXIMUM COOLANT TEMPERATURE (°F)	TIME AFTER REACTOR SHUTDOWN (HOURS)	TOTAL DECAY HEAT (MBTU/HR)
1a	161.2	185	19.8
1b	133.0	177	20.8
2	153.4	197	32.8
3	153.4	197	32.8

2.4.4 SFP Bulk Coolant Temperature

The license reported the maximum SFP bulk coolant temperatures to be as shown above in Table 1 for the four cases studied.

2.4.5 Maximum Spent Fuel Cladding Temperature

For conservatism, the licensee calculated the maximum cladding temperature for a spent fuel element as if it were operating at normal power in the core, 57.8 MBTU/hr, although assuming it to be in the SFP. The analysis utilized the fuel pool conditions for case 1a with both no channel blockage and 50% blockage, for further conservatism. The results are shown below (Table 2):

TABLE 2

MAXIMUM LOCAL POOL WATER AND FUEL CLADDING TEMPERATURE FOR THE LIMITING CASE (Case 1a)

	MAXIMUM LOCAL POOL WATER TEMP. °F	MAXIMUM LOCAL FUEL CLADDING TEMP. °F
No Blockage	230.5	256.4
50% Blockage	239.5	263.0

2.4.6 Time To Boil

The licensee calculated the time it would take to boil in the four cases. The calculation assumed that all forced cooling ceased at the instant the bulk coolant temperature reached its highest point. Results are shown below (Table 3):

TABLE 3

TIME TO BOIL

CASE	TIME TO BOIL (HOURS)	MAXIMUM EVAPORATION RATE (GPM)
1a	5.2	42.7
1b	7.9	43.0
2	3.6	70.7
3	3.6	70.0

2.4.7 Makeup Water

The plant has an emergency supply of water from the engine driven fire pump and river water pumps through seismic Class 1 piping. In addition water can be provided through other means by connections to the SFP cooling system. The approximate flow rates, the time required to initiate each water supply, and the power sources are included in Table 4 below:

TABLE 4

SFP MAKEUP WATER SOURCES

	MINIMUM FLOW	TIME TO PLACE IN SERVICE	ONSITE POWER
Primary Grade Water	50-100 gpm	Within 30 minutes	No
Refueling Water Storage Tank Cooling Systems	50-100 gpm	Within 30 minutes	No
River Water Seismic Class 1 Lines	300-400 gpm	Within 30 minutes	Yes
Engine Driven Fire Pump	3 hoses @ 100 gpm each	Within 30 minutes	N/A
Return of Fuel Building Air Conditioning to Fuel Pool	Insignificant in short term	Within 30 minutes	No

2.4.8 Summary

The staff conducted a calculation of the decay heat generation in comparison with that done by the licensee and found the licensee's results to be acceptable. The licensee calculated that the SFP bulk coolant temperature could reach 161 °F with the SFP containing 1524 spent fuel assemblies (including a refueling offload) and only one cooling train operating (Case 1a). This temperature exceeds that specified in the applicable guidelines (140 °F).

The plant has an operating procedure in place, AAD-1, "Spent Fuel Pool Temperature High," wherein the licensee takes action in the event the SFP coolant temperature reaches 143 °F. At that point an alarm is actuated and the plant personnel take the following corrective actions:

(1) Verify alarm by checking SFP temperature indicator.

(2) Verify component cooling water* temperature indicator.

(3) If fuel pool temperature is high and component cooling water temperature is in normal operating range, increase cooling water flow to operating heat exchanger.

(4) If cooling flow is at maximum, give consideration to placing second pump and heat exchanger in service.**

(5) If high temperature is a result of loss of component cooling water flow, consider connecting river water system to heat exchangers.

* Water used to cool SFP heat exchangers

** Crossties permit use of either pump with each heat exchanger

It is noted that the high temperature would occur only in the following circumstances: High component cooling water temperature (100 °F), failure of a complete SFP cooling train at or near the time the plant has unloaded a complete refueling load of spent fuel elements into the SFP and the inability of the plant to utilize any mitigating feature of procedure AAD-1. Further, 161 °F would be the highest temperature the bulk coolant would reach. It is estimated that the SFP bulk coolant temperature would be reduced by the operating SFP cooling train in 3 - 4 weeks to a level acceptable for long-term operation.

In view of the foregoing, the SFP coolant temperature for a normal refueling offload, with one operating SFP cooling train, is found to be acceptable.

The details of the plant operating procedure were clarified in telephone discussions between members of the staff of the Beaver Valley Plant and the NRC staff during August and September 1993.

The licensee calculated the maximum bulk coolant temperature to be 153.4 °F in the event a full core has to be unloaded. This temperature is in accordance with applicable guidelines and, thus, found to be acceptable.

The licensee calculated the cladding temperature to be expected, using Case 1a as the starting point. The cladding temperatures calculated conservatively provide results (256 °F and 263 °F) found to be acceptable by the staff. The licensee reported that those cladding temperatures produced no bulk nucleate boiling; thus, no coolant voids and no effect upon criticality beyond the normal density changes which are accounted for.

The licensee's calculations of times to boil, ranging from 3.6 to 7.9 hours, shows that there is sufficient time for operators to react should the SFP cooling system cease to function abruptly. The calculation is considered to be acceptable since the calculation assumed as starting points the maximum bulk coolant temperatures in the cases considered.

Makeup water, as shown in Table 4, may be supplied from five sources. All five systems can be employed expeditiously; the four with flow rates of 50 gpm or greater should suffice to maintain the SFP coolant level. Thus, the makeup systems are found to be acceptable.

In summary, the staff finds the licensee's plans regarding thermal/hydraulic aspects to be acceptable.

2.5 Mechanical and Seismic Loading

The existing SFP racks are bolted to the floor and braced to the walls, but the new racks would be free standing. The licensee proposed to replace the existing racks with the new free standing racks. However, there would be no alteration to the concrete SFP.

SFP racks are manufactured into a module, and each module is supported by a minimum of four legs. All rack modules are freestanding on the pool and modules are not connected to each other either. Since the new rack modules are free standing on the pool, the response of rack modules during earthquakes is complex because rack modules can slide, rock, twist, and overturn. Since the racks are under the water, the response of rack modules during earthquakes is complicated by the fluid coupling phenomena between adjacent racks and between racks and concrete walls. The response of rack modules during earthquakes is further complicated by the momentum transfers between fuel assemblies and cells resulting from the rattling of the fuel assemblies inside the storage cells. Therefore, to predict the response of rack modules during earthquakes by analysis is a difficult task.

The licensee contracted to Holtec International to perform 3-dimensional single rack analyses and a whole pool multi-rack analysis subjected to a safe shutdown earthquake event. The analyses were quite extensive and appeared to be performed carefully. The analyses results indicate: (1) that there is no rack-to-rack and rack-to-wall impact, because the displacements are small, (2) that the stresses at various sections of racks were far below the allowable stresses of the ASME Code, Section III, Subsection NF, (3) that the highest impact load between the fuel assembly and cell wall is 425 pounds which is much less than the allowable load of 3398 pounds based on permissible stresses, (4) that the calculated weld stresses between rack and baseplate and between pedestal and baseplate, and cell-to-cell are all far below the allowable stresses of welds, and (5) displacements of racks are small enough so that there is no possibility for overturning. The maximum displacement at the top of the rack is 0.31 in. calculated by the single rack analysis and 0.31 in. calculated by the whole pool multi-rack analysis. The maximum vertical load on pedestals is 170,170 pounds calculated by the single rack analysis and 133,600 pounds calculated by the whole pool multi-rack analysis. Displacements from both analyses are in good agreement and vertical loads on pedestals are not too far apart, and this is an indication that the whole pool multi-rack analysis has yielded reasonable results. The licensee stated that the hydrodynamic effect formulation for the multi-rack analyses was reviewed and approved previously by the NRC for Diablo Canyon Units 1 and 2, and the staff has confirmed this statement. The licensee also stated that laboratory experiments were conducted to validate the multi-rack fluid coupling theory used in the computer code DYNARACK which was used to analyze the response of the racks subjected to earthquakes. The licensee reported that test data of 199 stainless steel plates submerged in water show a mean value of coefficient of friction to be 0.503 with a standard deviation of 0.13. The licensee has used bounding values of 0.2 and 0.8 for Beaver Valley rack analyses. The staff believes that these bounding values are conservative. This information does offer evidence that the rack analyses performed for Beaver Valley Unit 1 have been extensive and have used conservative assumptions.

The Beaver Valley Unit 1 SPF is approximately 30 feet by 40 feet in dimensions. The structure is a seismic Category I reinforced concrete walled pool, located outside the containment structure, partially below grade. The interior of the pool is lined with a 1/4 inch thick stainless steel plate. Since the storage cells would be increased from 833 to 1627 (including two cans for defective fuel) as a result of reracking, the dead weight of the racks including fuel load on the SFP and the soil beneath the pool would also increase. The licensee reported that the maximum calculated average pressure on concrete in the vicinity of highly loaded pedestals is 1182 psi which is less than the allowable stress of 3570 psi. The licensee also reported that the foundation pressure on soils is 6457 psf in static condition, which is less than the allowable stress of 8,000 psf, and 8394 psf with seismic effect, which is less than the allowable stress of 12,000 psf. Therefore, the staff believes that the licensee has demonstrated that the SFP and the soils that support the pool are capable of taking additional loads resulting from the reracking.

Based on the review results of the licensee's submittals and a discussion with the staff of Holtec International, the NRC staff has determined that the proposed changes for reracking are acceptable. This is because the staff has concluded that the licensee has performed extensive analyses and employed conservative assumptions for the analyses. The staff further believes that the validation of fluid coupling theory for the computer code by laboratory tests has offered some confidence to the adequacy of the computer code. The analyses results have indicated that the rack displacements are small and there is no rack-to-rack or rack-to-wall impacts and the stresses in the fuel assemblies, cell walls, welds, and racks are far below their allowable stresses. The Beaver Valley plant is located in a low seismic zone, and this adds additional comfort to the judgment of the adequacy of the reracking for Beaver Valley Unit 1.

2.6 Radiation Protection

2.6.1 Occupational Dose Control

The licensee estimated in its November 2, 1992, application that total occupational dose for planned reracking activities would be between 6 and 12 person-rem, including any necessary diving operations.

This overall estimate is based on individual dose estimates for each of the series of anticipated activities to be performed during the reracking operation. These activities include removing and decontaminating (hydrolasing) the current racks once they are emptied and removed from the fuel pool; removing underwater appurtenances; installing new racks; and preparing the old racks for shipment.

The licensee has indicated that the removal of underwater appurtenances will be performed using remote handling tools to the greatest extent possible. If diving operations are required, careful monitoring and adherence to procedures should assure that the radiation dose to the divers is ALARA. Further, if divers are used, the licensee has substantially committed to the guidance provided in Appendix A ("Procedures for Diving Operations in High and Very High Radiation Areas") to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants." The licensee's proposed departure from the guidance in Appendix A to Regulatory Guide 8.38, consisting of the use of remote indicating instrumentation for diving operations coupled with continuous communication between the diver and the individual monitoring the remote indications, is acceptable as a substitute for the use of alarm dosimeters by divers.

The licensee notes that detailed procedures prepared with consideration of the ALARA principle will be utilized. In addition, DLC states in its submittal that continuous air samplers will be utilized where a potential for significant airborne activity exists and that personnel will wear protective clothing and, as appropriate, respiratory protective

equipment. Further, work activities are to be governed by Radiation Work Permits specifying appropriate radiation protection measures. In addition to the routine use of self-reading dosimeters and thermoluminescence dosimeters, extremity badges and alarm dosimeters will be utilized as appropriate. The licensee further states that work activities, personnel traffic, and equipment movement will be monitored and controlled such that contamination will be minimized and personnel exposures maintained ALARA. Based on our review of the licensee's application, the staff finds the proposed radiation protection aspects of the SFP rerack acceptable.

2.6.2 Solid Radioactive Waste

The licensee stated in its application that the existing spent fuel storage racks will be ultimately shipped to a licensed disposal site as low level radioactive waste. Estimates of the person-rem exposures associated with washdown and preparation of the old racks for shipment were made and found to be acceptable. Shipping containers and procedures will conform to Department of Transportation (DOT) regulations and the requirements of the state through which the shipment may pass, as set forth by the State DOT office.

Based on our review, the staff finds that the licensee's plan for handling and disposing of solid radioactive waste generated in connection with the planned reracking operation meets regulatory requirements and is, therefore, acceptable.

2.6.3 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of postulated accidents, included means for their avoidance in the design and the operation of the facility, and provided means for mitigation of these consequences should they occur. The licensee has evaluated the effect and the calculated consequences of a spectrum of postulated design basis accidents (DBA) and concludes that the effects of the proposed TS changes are small and that the calculated consequences are within regulatory requirements and staff guideline dose values. Since the licensee proposes to utilize higher enrichment fuel, the staff reevaluated the postulated fuel handling DBA for Beaver Valley to consider the effects of increased enrichment and burnups.

In its evaluation for Beaver Valley, issued on October 1985, the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from the fuel handling DBA. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in NUREG-0800.

Since the applicant intends to utilize higher enrichment fuel, the staff has reanalyzed the fuel handling DBA for this case. According to NUREG/CR-5009 (February 1989), increasing fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T could increase the calculated doses for a fuel handling accident by a factor of 1.2. The licensee proposes to limit

enrichment to 5.0 weight percent U-235 with a maximum burnup of 40,000 MWD/T. Therefore, the 1.2 factor increase in dose displayed in Table 1 below, bounds the dose consequences of the licensee's proposal. In Table 1, the new and old DBA doses are presented and compared to the guidelines doses in NUREG-0800 (based on 10 CFR Part 100).

TABLE 1
Radiological Consequences of Fuel
Handling Design Basis Accident (REM)

	Exclusion Area	Low Population Zone
	Thyroid	Thyroid
Staff Evaluation October 1985	25	1.0
Bounding Estimates for Higher Enrichment Fuel Burnup*	30	1.2
Regulatory Requirement (NUREG-800 Chapter 15.7.4)	75	75.0

* Factor of 1.2 greater than original estimate for iodine.

The staff has determined that the only fuel handling accident dose that may be increased due to the increased enrichment is the thyroid dose and that this dose remains well within the dose limits set forth in NUREG-0800 and is, therefore, acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental asssessment and finding of no significant impact have been prepared and published in the Federal Register on October 23, 1993 (58 FR 58201). Accordingly, based upon environmental assessment, the Commission has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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