March 17, 19--

Docket Nos. 50-454 and 50-455

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Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: INCREASE IN THE SPENT FUEL POOL CAPACITY THROUGH THE USE OF HIGH DENSITY STORAGE RACKS, BYRON STATION, UNITS 1 AND 2 (TAC NOS. 62112 AND 63266)

By letter dated September 3, 1986, supplemented November 7 and November 24, 1986, Commonwealth Edison Company (the licensee) requested approval to increase the capacity of the spent fuel pool through the use of high density storage racks.

We have reviewed the initial request, its supporting information, and answers to the staff questions contained in subsequent letters; and find the proposed changes acceptable. The amendment authorizes the licensee to increase the capacity of the Byron Station spent fuel pool from the currently approved capacity of 1060 fuel assemblies to the proposed capacity of 2870 fuel assemblies.

It was noted during the staff review that while the proposed surveillance program for monitoring the Boraflex in the spent fuel pool was acceptable, no corrective action was proposed in the event that Boraflex degradation was observed. It is recommended that a plan of corrective actions be developed and implemented.

Our Safety Evaluation is enclosed.

Sincerely,

/S/

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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

March 17, 1989

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Enclosure:
As stated

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See next page

Mr. Henry E. Bliss
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Byron Station
Units 1 and 2

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-454

BYRON STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-37

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 3, 1986, supplemented November 7 and November 24, 1986, 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 indicted in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 25 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-455

BYRON STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-66

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 3, 1986, supplemented November 7 and November 24, 1986, 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-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 25 and revised by Attachment 2 to NPF-60, and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

ATTACHMENT TO LICENSE AMENDMENT NOS. 25 AND 25

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. 50-454 AND 50-455

Revise Appendix A as follows:

Remove Pages	Insert Pages
5-5	5-5 5-5a

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in Section 9.1 of the FSAR. This is based on spent fuel storage in Region 2 with enrichments and burnup in accordance with Figure 5.6-1 or in a checkerboard pattern; and

b. A nominal 10.32 inch north-south and 10.42 inch east-west, center-to-center distance between fuel assemblies placed in Region 1 spent fuel storage racks and a nominal 9.03 inch center-to-center distance between fuel assemblies placed in Region 2 spent fuel storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 423 feet 2 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2870 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

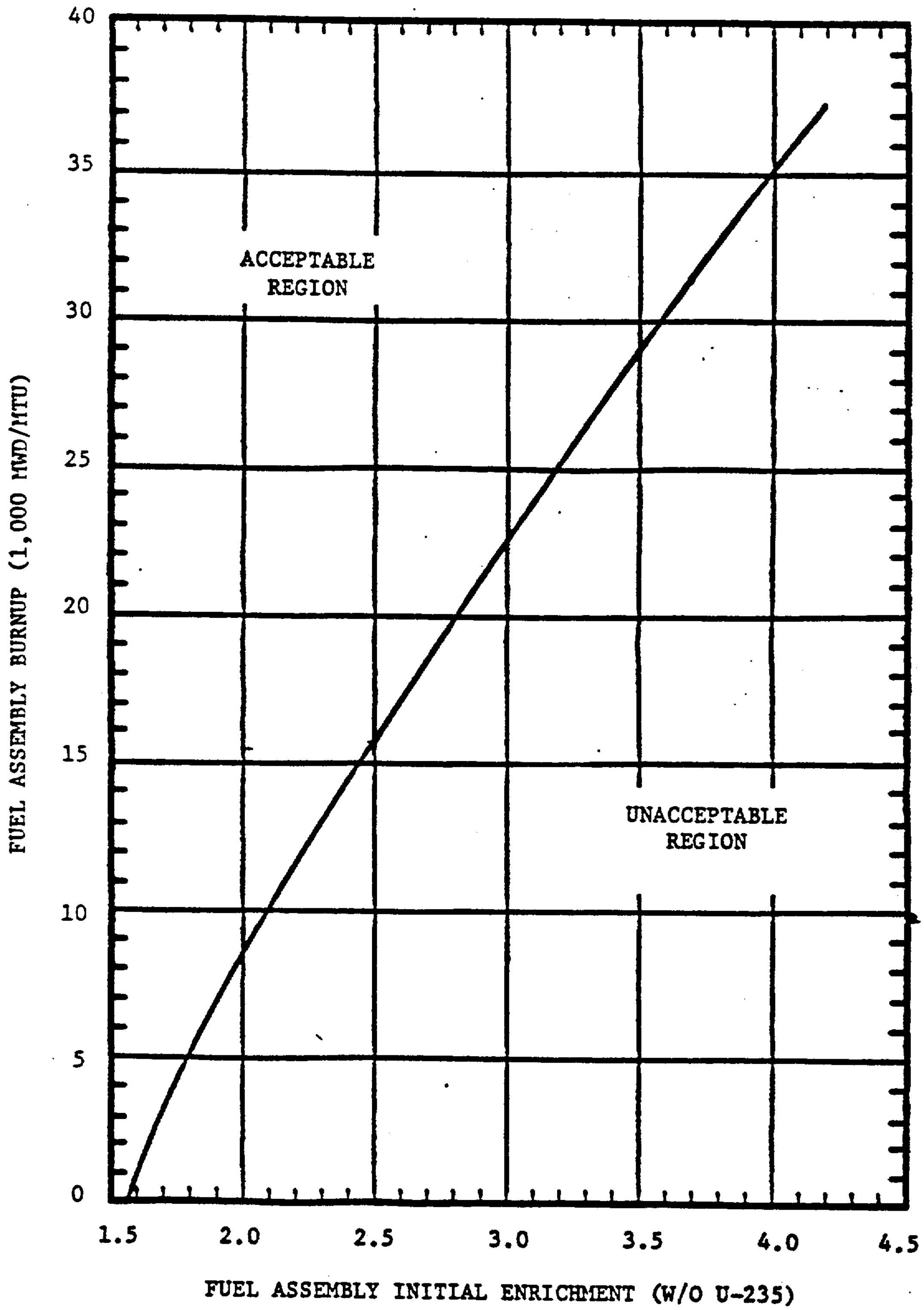

FIGURE 5.6-1

MINIMUM BURNUP VERSUS INITIAL ENRICHMENT
FOR REGION 2 STORAGE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE INCREASE IN THE SPENT FUEL POOL CAPACITY

THROUGH THE USE OF HIGH DENSITY STORAGE RACKS

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-454 AND 50-455

BYRON STATION, UNITS 1 AND 2

1. INTRODUCTION

1.1 Submittal and Staff Review

This report presents the NRC staff's Safety Evaluation for the reracking of the spent fuel pool at Byron Station. By letter dated September 3, 1986, supplemented November 7 and November 24, 1986, Commonwealth Edison Company (the licensee) submitted a request to increase the storage capacity of the spent fuel pool.

The request is based on the licensee's "Licensing Report on High Density Spent Fuel Racks for Byron Station, Units 1 and 2," which was submitted as an enclosure to the September 3, 1986 letter. During its review, the staff requested additional information which was provided by letters dated December 11, 1986; March 11 and December 22, 1987; May 26, June 1, and August 17, 1988.

1.2 Summary Description of Reracking

A single spent fuel storage facility located in the fuel handling building is shared by both Units 1 and 2 at Byron Station. The facility includes the spent fuel storage racks and the stainless-steel-lined spent fuel storage pool that contains the storage racks.

The licensee requested approval to increase the spent fuel pool storage capacity from the previously approved number of 1060 to 2870 fuel assemblies. The proposed expansion is to be achieved by reracking the spent fuel pool into two discrete regions. New, high-density storage racks (free-standing) will be used.

The high density spent fuel racks consist of individual cells with 8.85-inch (nominal) square cross-section, each of which accommodates a single Westinghouse PWR fuel assembly or equivalent. A total of 2864 cells and six failed fuel storage cells are arranged in 23 distinct modules of varying sizes in two regions. Region 1 is designed for storage of new fuel assemblies with enrichments up to 4.2 weight percent U-235. Region 1 is also designed to store fuel assemblies with enrichments up to 4.2 weight percent U-235 that have not achieved adequate burnup for Region 2. The Region 2 cells are capable of accommodating fuel assemblies with various initial enrichments which have accumulated minimum burnups within an acceptable bound as depicted in proposed Technical Specification Figure 5.6-1.

The rack module is fabricated from ASTM A-240-304L austenitic stainless steel sheet and plate material, ASME SA351-CF3 and SA217-CA15 casting material and SA 479-410 material. The weld filler material utilized in body welds is ASME SFA-5.9, Type 308L and 308LSI. Boraflex and Boral serve as the neutron absorber material.

The new racks are not doubled-tiered and all racks will sit on the spent fuel pool floor.

The proposed expansion of the spent fuel pool storage capacity to 2870 fuel assemblies should provide adequate storage until the year 2009, while maintaining full core offload capability. In addition, the expansion should be adequate until a federal repository is available for spent fuel.

2 CRITICALITY CONSIDERATIONS

2.1 Criticality Analysis

2.1.1 Calculations Methods

The primary method of analyses used for the Byron racks was the CASMO-2E code. This is a two dimensional multigroup transport theory code that is capable of an accurate representation of the rack geometry. This code is routinely used for calculating small reactivity increments for evaluating uncertainties and has also been previously used for primary calculations. The code has been extensively verified for calculation of core reactivities and the present submittal presents the results of a verification of CASMO-2E against KENO (Monte-Carlo) calculations for critical experiments which mock up spent fuel rack geometries. These comparisons show that the CASMO-2E results are well within the statistical uncertainty in the Monte Carlo results. We conclude that the CASMO-2E code is acceptable for the analysis of spent fuel racks.

For additional verification of the method, the nominal cases for Regions 1 and 2 were calculated with the KENO code with the AMPX-NITAWL cross-section preparation program set. This is the industry standard code for this purpose and the submittal presents the results of its verification against critical experiments. Also the nominal case for Region 1 was calculated by a diffusion theory code. The results showed excellent agreement among the codes, providing additional assurance that the methods of calculations are appropriate.

The staff concludes that acceptable calculational methods have been used for the criticality analysis of the racks. The CASMO-2E code is also used for the calculation of the burnup effects on the reactivity of the assemblies. This code is widely used in the industry for this purpose. The submittal presents comparisons of the results of the reactivity as a function of burnup among three standard codes for two different enrichment bundles. The three codes are in excellent agreement with the CASMO code, generally giving results between the other two. The staff concludes that the CASMO code is acceptable for the calculation of burnup effects.

2.1.2 Calculations and Uncertainties

2.1.2.1 Assumptions Used

In order to provide conservatism in the results and to simplify the calculations, the following assumptions were made in obtaining the nominal values of the k-effective of the pool:

- The moderator is unborated water at the pool temperature corresponding to the highest reactivity.
- The storage racks are assumed to be infinite in extent in both the vertical and lateral dimension.
- The absorption effect of minor structural components of the assemblies (e.g., spacer grids) is neglected.

These are the traditional assumptions for such calculations and are acceptable.

2.1.2.2 Consideration of Uncertainties

The nominal value of the k-effective of the racks is determined by using nominal values for each of the dimensions, enrichments, densities, etc. which are input for the calculation. Calculations are performed with changed values of each parameter and the sensitivity of the results to that parameter is obtained. The uncertainty in the k-effective value due to that parameter may be obtained by forming the product of the sensitivity and the uncertainty in the parameter. The total uncertainty is then obtained by combining the uncertainties due to all the parameters.

For the present analysis, a calculational bias and uncertainty in that bias were obtained from the comparison of the calculation methods with critical experiments. Uncertainties due to tolerance variations were obtained for the following parameters:

- Boron-10 concentration
- Boraflex thickness
- Boraflex width
- Inner box dimension
- Water gap thickness
- Box wall thickness
- Fuel enrichment
- Fuel density
- Eccentric assembly position

In addition to these tolerance uncertainties, an uncertainty in the k-effective due to burnup uncertainty was determined. This uncertainty was conservatively treated as a bias in the evaluation of the Region 2 racks.

The staff concludes that appropriate uncertainties have been considered and that the treatment of uncertainties is acceptable.

2.1.3 Results of the Calculation

The nominal values for the k-effective of the racks are 0.9374 for Region 1 and 0.8999 for Region 2. The calculational bias is 0.0013 for both regions and the values of the total uncertainties at the 95 percent probability, 95 percent confidence level are 0.0082 and 0.0093 for Regions 1 and 2, respectively. For Region 2, an additional allowance for burnup uncertainty of 0.0187 is added. Where these values have been combined, the resulting k-effectives, including all uncertainties and biases, are 0.9469 for Region 1 and 0.9292 for Region 2. These values meet our criterion of 0.95 for this quantity and are acceptable.

2.1.4 Abnormal and Accident Conditions

Several abnormal and accident conditions have been considered for the racks. These include increase in pool water temperature, boiling (void formation in the pool), dropping a fuel assembly on top of the racks, lateral movement of the racks (seismic event) and misloading of a Region 1 assembly into Region 2. Only the last of these has the potential for a significant increase in pool reactivity. In case of a misloaded assembly, credit may be taken for the presence of boron in the pool water. The boron is required by the Technical Specifications and periodic surveillance of the pool boron concentration is performed. Calculations show that only 300 parts per million (ppm) boron is required to assure that the Region 2 racks meet the acceptance criterion for k-effective if a fresh fuel assembly is present. The presence of 2000 ppm of boron assures that the criterion will be met if all Region 2 locations are filled with fresh fuel. The staff concludes that the analysis of abnormal conditions is acceptable.

2.1.5 Dry Storage of Fresh Fuel in Spent Fuel Racks

Region 1 is designed for safe storage of fresh fuel at all moderator densities from dry conditions to full density unborated water. It may be desirable also to use Region 2 racks for dry or wet temporary storage of fresh fuel. Analyses were performed to show that a checkerboard pattern (fuel assemblies aligned diagonally) provides an acceptable k-effective in either the fully flooded or dry (low density moderation) conditions.

The staff concludes that temporary storage of fresh fuel in Region 2 in a checkerboard pattern is acceptable.

2.1.6 Technical Specifications

Specification 5.6.3 is altered to increase the storage capacity of the spent fuel pool to 2870 fuel assemblies. This value is consistent with that in the submittal and is acceptable.

Proposed Specification 5.6.1.1a refers to the FSAR for a description of the analysis methods and uncertainties. The FSAR will be updated to be consistent with the methods and uncertainties of the present analyses. This is acceptable. This Specification also references Figure 5.6-1, which will be added to the Specifications, for the curve of minimum burnup as a function of initial enrichment required for storage in Region 2. This curve is consistent with the analyses and is acceptable.

Proposed Specification 5.6.1.1.b is a description of the proposed racks. It is consistent with the submittal and is acceptable.

2.1.7 Conclusions

Based upon the review which is described above, the staff finds the criticality aspects of the proposed reracking of the Byron spent fuel pool to be acceptable. This conclusion is based on the following:

1. The analyses were performed by acceptable methods which have been verified by comparison with experiment.

2. Acceptable assumptions have been made with respect to the input values of the parameters.

3. Appropriate uncertainties have been treated with acceptable methods.

4. Acceptable analysis of abnormal and accident conditions have been performed.

5. The k-effective values, including all uncertainties, meet our criterion of less than or equal to 0.95 for this quantity.

6. The analyses show that fresh fuel may be stored in Region 2 provided a checkerboard arrangement is used.

7. The proposed Technical Specifications are consistent with the submittal and are acceptable.

3 MATERIAL COMPATABILITY AND CHEMICAL STABILITY

3.1 Discussion

Nuclear reactor plants provide storage facilities or pools for the wet storage of spent fuel assemblies. The safety function of the spent fuel storage pools is to maintain the spent fuel assemblies in a sub-critical array during all credible storage conditions. The staff has reviewed the compatability and chemical stability of the materials (except the fuel assemblies) wetted by the pool water, in accordance with Section 9.1.2 of the Standard Review Plan (NUREG-0800, July 1981).

The spent fuel storage pool at the Byron Station, Unit 1 and 2, contains oxygen saturated demineralized water which has a nominal concentration of 2000 parts per million (ppm) of boron. The pool is lined with stainless steel. The principal construction materials for the proposed new racks in the spent fuel storage pool are ASTM A-240 Type 304L austenitic stainless steel for structure, and Boraflex and Boral for neutron absorption. The racks are interconnected honeycomb arrays of square stainless steel boxes forming individual cells for fuel storage.

The spent fuel pool consists of two regions: Region 1 is designed to store new fuel assemblies and fuel assemblies that have not achieved adequate burnup; Region 2 is designed to store fuel assemblies that have achieved adequate burnups. In Region 1, Boraflex sheets are placed on all four sides of the square cells over the active fuel length except the top and bottom 2.25 inches. In addition to the Boraflex, Boral sheets are added to the flux traps of two racks which have already been built. Boral sheets will be used in place of Boraflex to cover the entire active fuel length on racks that are not yet built. The Boraflex sheets are covered by stainless steel plates which are spot welded onto the square boxes. No cover plates are used for the Boral sheets.

In Region 2, a Boraflex sheet is placed between two adjacent square cells. Stainless steel strips are inserted on both sides and bottom of the Boraflex sheet to envelope the entire active fuel length. The two adjacent cells and the side strips between them are welded together.

In both Regions 1 and 2, the Borafex sheets are not physically fastened onto any surface and no adhesive was used on the Boraflex panels.

The licensee has developed an inservice surveillance program to monitor the performance of the Boraflex. Twenty test coupons that are representative of the Boraflex sheets are placed in each region of the spent fuel pool. One of these coupons is removed from each region every other year to be examined for physical stability, hardness, neutron radiography, and neutron attenuation. In addition, after the first refueling, a blackness test will be performed on a representative sample of storage cells which temporarily had spent fuel assemblies stored in them to ensure acceptability for continued use. Should degradation of Boraflex be found, the licensee will make a criticality evaluation to ensure that the nuclear safety limits are maintained. No other corrective action is proposed.

3.2 Evaluation

The stainless steel in the storage pool liners and rack assemblies is compatible with the oxygen-saturated borated water and radiation environment of the spent fuel pool. In this environment, corrosion of Type 304L stainless steel is not expected to exceed a rate of 6×10^{-7} inch per year (E. G. Brush and W. L. Pearl, "Corrosion and Corrosion Product Release in Neutral Feedwater," Corrosion, Vol. 28, p. 129, April 1972). This corrosion rate is negligible for even the thinnest stainless steel walls in the rack assemblies. Contact corrosion or galvanic attack between the stainless steel in the pool liners or rack assemblies and the Inconel/Zircaloy in the fuel assemblies to be stored will not be significant because all these materials are protected by passivating oxide films. Boraflex is composed of non-metallic materials and, therefore, will not develop a galvanic potential with the metal components.

Space is available to allow escape of any gas which may be generated from the polymer binders in the Boraflex during heating and irradiation, thus preventing possible bulging or swelling of the Boraflex assemblies. Boraflex, an elastomer of methylated polysiloxane filled with boron carbide power, is used as a neutron absorber (poison) in the spent fuel storage facilities of many nuclear power plants. It has undergone extensive testing to determine the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron absorbing material (Bisco Products, Inc., Technical Report No. NS-1-001, "Irradiation Study of Boraflex Neutron Shielding Materials," August 12, 1981). The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex in 2000 ppm boron solution to 1.03×10^{11} rads of gamma radiation with a concurrent neutron flux of 8.3×10^{13} neutrons/cm^2/sec. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma and neutron irradiation. However, irradiation caused some loss of flexibility and shrinkage of the Boraflex.

Long-term borated water soak tests at high temperatures were also conducted (Bisco Products, Inc., Technical Report No. NS-1-002, "Boraflex Neutron Shielding Material Product Performance Data," August 25, 1981). The tests show that Boraflex withstands a temperature of 240°F in a solution of 3000 ppm boron for 251 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide. The spent fuel pool water temperature under normal operating conditions will be approximately 105°F which is well below the 240°F test temperature. In general, the rate of a chemical reaction decreases exponentially with decreasing temperature. Therefore, the staff does not anticipate any significant deterioration of the Boraflex at the normal operating conditions of the pool over the design life of the spent fuel racks.

The tests have shown that neither irradiation, environment, nor Boraflex composition have a discernible effect on the neutron transmission of the Boraflex material. The tests also have shown that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally present in the Boraflex will typically contain 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation capability of the silicone polymer matrix to prevent the leaching of soluble species from the boron carbide.

Recently, anomalies ranging from minor physical changes in color, size, hardness, and brittleness to formation of gaps up to four inches wide were observed in Boraflex panels that have been used in the spent fuel pools of three nuclear power plants. The exact mechanisms that caused the observed physical degradations of Boraflex have not been confirmed. The staff postulates that gamma radiation from the spent fuel initially induced crosslinking of the polymer in Boraflex, producing shrinkage of the Boraflex material. When crosslinking became saturated, scissioning (a process in which bounds between atoms are broken) of the polymer predominated as the accumulated radiation dose increased. Scissioning produced porosity which allowed the spent fuel pool water to permeate the Boraflex material. Scissioning and water permeation could embrittle the Boraflex material. In short, gamma radiation from spent fuel is the most probable cause of the observed physical degradations, such as changes in color, size, hardness, and brittleness. The staff does not have sufficient information to determine conclusively what caused the gap formation in some Boraflex panels. However, it is conceivable that if the two ends of a full-length Boraflex panel are physically restrained, then shrinkage caused by gamma radiation can break up the panel and lead to gap formation.

The staff determined that reasonable assurance exists that physical restraints are absent in the Boraflex panels of the Byron Station, because the Boraflex sheets are not physically fastened to or permanently glued onto any structure. It is not likely that gaps will form in any significant extent in the Boraflex panels during the projected life of the Boraflex assemblies. However, minor physical degradations can take place in the Boraflex from irradiation.

In the unlikely event of gap formation in the Boraflex panels that would lead to loss of neutron absorbing capability, the monitoring program will detect such degraded Boraflex panels, and the licensee would have sufficient time to perform a criticality evaluation.

Boral has also been used as a neutron absorbing material in the spent fuel storage pools of many nuclear power plants. Boral sheets consist of a baked matrix of boron carbide and aluminum type 1100 alloy, cladded on both sides by aluminum type 1100 alloy. The nominal thicknesses of the Boral sheets used at the Byron Station are 0.075 and 0.085 inch. The aluminum cladding prevents direct contact of the matrix with water in the spent fuel pool, except for the outer edges of the Boral panels.

The wettable amount of boron carbide matrix at the outer edges of Boral sheets is less than one percent of the total boron carbide contained therein (Brooks and Perkins, Inc., Report 624, "BORAL Product Performance," 1987). The boron carbide in Boral is allowed to contain, by the ASTM specification C750-80, up to three percent soluble boron oxide. Thus, the maximum leachability of boron carbide is 0.03 percent. This leachability would not significantly degrade the overall physical integrity of Boral sheets. Tests conducted at the University of Michigan showed no leachable halogen from irradiated Boral.

The general corrosion rate of aluminum similar to type 1100 alloy in water of pH 7 at a temperature of 125°C (257°F) has been measured to be 1.5×10^{-4} mils per day or 2.2 mils in 40 years (J. E. Draley and W. E. Ruther, Argonne National Laboratory ANL-5001, February 1953). The weight loss rate due to galvanic corrosion of aluminum coupled with stainless steel type 304 in water of pH 5.0 at a temperature of 100°C (212°F) was determined to be 0.1 to 0.2 mil per year (Brooks and Perkins, Inc., Report 624, "BORAL Product Performance," 1987). Such corrosion rates for the aluminum in Boral are negligible for the designed lifetime of the spent fuel pool.

Irradiation of Boral plates in dry air, distilled water, and a solution of 2000 ppm boron at a gamma radiation flux of 7×10^{7} rad per hour, thermal neutron flux of 1.1×10^{13} neutrons/cm^2-sec, and fast neutron flux of 1.1×10^{11}

neutrons/cm^2-sec for up to 14290 hours, showed no detectable gas evolution from the Boral (R. R. Burn and G. Blessing, Transactions of the American Nuclear Society, Volume 32, Supplement 1, pp. 48-49, 1979). Irradiation of Boral matrix material with a cumulative exposure of 4×10^{16} thermal neutrons/cm^2, 1.2×10^{14} epithermal neutrons/cm^2, 5.4×10^{15} fast neutrons/cm^2, and 1.5×10^9 rad gamma rays resulted in no detectable gas evolution (Brooks and Perkins, Inc., Report 578, "The Suitability of Brooks & Perkins Spent Fuel Storage Module for Use in PWR Storage Pool," July 7, 1978). Calculations (ibid) of helium generation from nuclear reaction of boron-10 with neutrons in a typical Boral matrix indicated a potential pressure rise of 4.6×10^{-5} atmosphere (7×10^{-3} pounds per square inch) over a period of 40 years. Such a pressure build-up is insignificant and should not affect the physical integrity of the Boral sheets.

3.3 Conclusions

Based on the above discussion, the staff concludes that the corrosion of the spent fuel pool components due to the pool environment should be of little significance during the life of the facility. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion.

The staff further concludes that the environmental compatability of the materials used in the spent fuel storage pool is adequate based on the test data cited above and actual service experience at operating reactor facilities.

The staff has reviewed the proposed surveillance program for monitoring the Boraflex in the spent fuel storage pool and concludes that the program can reveal deterioration that might lead to loss of neutron absorbing capability during the life of the spent fuel storage racks. However, if a significant loss of neutron absorbing capability is found in any Boraflex panel, the licensee should take corrective actions, such as replacement of the rack module having the degraded Boraflex panel or restriction of use of the affected cell for fuel storage.

The staff finds that the proposed monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR 50, Appendix A, General Design Criterion 61 regarding the capability to permit appropriate periodic inspection and testing of components, and General Design Criterion 62 regarding prevention of criticality by the use of boron poison and by maintaining structural integrity of components, and are, therefore, acceptable.

4 STRUCTURAL DESIGN

4.1 Introduction

This evaluation addresses the adequacy of the structural aspects of the proposed application. The Brookhaven National Laboratory (BNL) assisted the staff in reviewing various analyses and responses submitted by the licensee, and in auditing the methodology and sample calculations. Attached Appendix A is the technical evaluation report (TER) developed by BNL. The staff accepts the findings and conclusions of the TER by incorporating the TER as a part of this evaluation.

Two units of Byron Station share one common pool. The pool is 62 feet 0 inch long(N-S) and is 33 feet 1 inch wide(E-W), and is centrally located between the containments of the two units in the Fuel Handling Building (FHB). The reinforced concrete pool walls are 5 feet 0 inch to 6 feet 0 inch thick. The walls separating the cask pit and the spent fuel pool are 2 feet 6 inch thick. The pool floor is the top of the reinforced concrete basemat of the FHB resting directly on bedrock. The pool walls and floor are lined with 1/4 inch - 3/16 inch stainless steel liner plate to ensure the water tightness of the pool. The five 1.5 inch drain lines installed behind the liner would collect any leakages through the liner.

The proposed high-density storage racks consist of individual cells with 8.85 inches by 8.85 inches square cross-section, each of which would accommodate a single Westinghouse fuel assembly or equivalent. A total of 2870 cells are arranged in 23 distinct rack modules of various arrays of fuel cells. Each rack module is equipped with 1 inch thick by 3½ inch high girdle bars at the upper end designed to withstand the impact loads under the postulated seismic conditions. The rack modules are free-standing, and they make surface contact at the girdle bar locations providing a nominal 2 inch gap between adjacent module cell walls. A detailed description of the rack modules is provided in Appendix A.

The proposed application is for the storage of a single fuel assembly in each storage location of the high density racks. However, most of the analyses have been performed with the consolidated fuel weight in the storage locations. For the sake of analysis, however, the conservative assumptions have been made to simulate gaps and spring constants. The staff finds the approach acceptable for evaluating the proposed reracking.

4.2 Evaluation

The primary areas of review associated with the proposed application are focussed towards assuring the structural integrity of the fuel, fuel cells, rack modules, and the spent fuel pool floor and walls under the postulated loads (Appendix D of SRP 3.8.4) and fuel handling accidents. The major areas of concern and their resolutions are outlined in the following paragraphs.

4.2.1 Fuel Handling Building and Spent Fuel Storage Pool

The Fuel Handling Building analysis and design had been reviewed and accepted during the initial licensing stages. The pool floor slab and walls were reanalyzed to account for the added load of the fuel (consolidated), the racks and the associated impact loads. The pool slab is founded on rock. However, the reanalysis was performed using the supporting medium data from Braidwood site (i.e., softer medium) giving higher forces in the pool structures. The pool walls were analyzed for potential impact loads resulting from the seismic analysis of the free standing racks. The rack support legs will be resting on a minimum of 1 inch thick shim plates, thus distributing the potential impact loads. The stresses in concrete and reinforcing steel at critical sections are found to be within the acceptable criteria. A detailed evaluation of the affected spent fuel pool components is provided in Section 4.6 of Appendix A.

4.2.2 High Density Racks

The analysis of high density racks was performed using a three-dimensional model of a rack with simulation of masses, springs, gaps and support conditions. A number of analyses were performed to demonstrate the potential behavior of racks under the postulated seismic events (SSE and OBE). Because of the large (over 1 inch) displacements exhibited in the edge and corner rack analysis, the staff questioned the adequacy of the single rack analysis. The licensee performed two-dimensional multirack analyses of one row of racks using bounding values of friction coefficients. These analyses showed lower rack to rack and fuel to rack impact loads, but they showed substantial rack to wall impacts.

Major components of the rack were evaluated for the maximum impact loads under the controlling load combinations; for example, the minimum ratio of applied load to the code allowable value is 1.24 for the weld between the baseplate and the support foot, and that for the fuel cell is 3.16. Table 6 of Appendix A provides the ratios (safety factors) associated with each of the major structural components of the rack.

The fuel rack system was also evaluated for the inadvertent drop of a fuel assembly during fuel handling operation. Two cases were analyzed: (1) with a fuel assembly dropping on the top of the rack and going through the height of the storage rack and hitting the base plate, (2) with a fuel assembly dropping from 36 inches above the rack. Energy balance approach with conservative assumptions indicated that in case (1) the base plate would not be perforated, and in case (2) the large plastic deformation would be limited to the rack module above the active fuel region. This is acceptable under this type of accident. A detailed evaluation of the dropped fuel accidents is provided in Section 4.5 of Appendix A.

4.2.3 Conclusion

Based on its evaluation of the licensee's submittal, the supplementary information provided by the licensee, discussion with the licensee at meetings, and information audited by the staff and its consultant, the staff concludes that the licensee's structural analyses of the spent fuel rack modules and the spent fuel pool are in compliance with the acceptance criteria set forth in the Standard Review Plan and consistent with the current licensing practice and, therefore, are acceptable. It should be pointed out, however, that the installation of new racks (wet reracking) on the existing embedded plates and additional shim plates will require utmost care in levelling and spacing the racks in the desired configuration. A thorough review of the installation procedures and inspection of the installed racks is warranted.

5 SPENT FUEL POOL COOLING AND LOAD HANDLING

5.1 Spent Fuel Pool Cooling System

The spent fuel pool cooling system consists of two independent trains each consisting of one pump and heat exchanger with associated piping and valves. The spent fuel pool cooling pumps can be powered from emergency (Class 1E) power sources. No modification to the spent fuel pool cooling system is proposed with this increase in storage capacity from 1060 to 2870 assemblies. The spent fuel pool cooling system was reviewed against the requirements of General Design Criterion (GDC) 44 for decay heat removal and GDC 2 for makeup during loss of all cooling according to Standard Review Plan (SRP), Section 9.1.3.

5.1.1 Decay Heat Removal

The licensee calculated the decay heat loads of the spent fuel assemblies discharged to the pool in accordance with the Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long Term Cooling," and Standard Review Plan (SRP) Section 9.1.3, "Spent Fuel Cooling and Cleanup System." The maximum normal heat load following the last refueling, 128 hours after shutdown, was calculated to be 35.91 MBTU/Hr. This heat load will result in a maximum bulk pool temperature of 138.5°F with one cooling train operating. These results meet the acceptance criterion of at or below 140°F for maximum normal heat loads as defined in the SRP Section 9.1.3 assuming single active failure. The expected maximum abnormal heat load following a full core discharge, 17 days after the normal discharge from the other unit, was calculated to be 56.66 MBTU/Hr. This abnormal heat load will result in a maximum bulk pool temperature of 158.1°F. It meets the acceptance criterion of below boiling for maximum abnormal heat loads as defined in the SRP Section 9.1.3.

The NRC staff performed an independent calculation which confirmed the applicant's heat load calculations. As a result of its review, the NRC staff finds that the spent fuel pool cooling system still meets the requirements of GDC 44 with respect to providing adequate cooling including single failure consideration.

5.1.2 Loss of Cooling

The licensee calculated that, assuming the loss of all cooling, boiling would occur after 9.0 hours for maximum normal heat load conditions and after 3.8 hours for maximum abnormal heat load conditions and will result in a boil-off rate of 71.6 and 116.8 gpm, respectively. This provides reasonable time to initiate makeup to the spent fuel pool from the seismic Category I refueling water storage tanks or the primary makeup system or the fire protection system.

Because the seismic Category I makeup source is more than adequate to provide water for the higher boil-off rate of the expanded storage capacity, the design still meets the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena."

Based on the above, the staff concludes that the spent fuel pool cooling system meets the requirements of SRP Section 9.1.3 and is therefore acceptable.

5.2 Heavy Load Handling

The new spent fuel racks are considered to be heavy loads as they weigh more than a spent-fuel assembly and its handling tool. The reracking of the spent fuel pool involves installation of 23 new high density racks and removal of 12 old racks using a 125 ton overhead fuel handling building crane which is designed to seismic Category I requirements. The licensee stated that the installation of high density spent fuel racks will be performed in accordance with NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36", Section 5.1.2, Alternative 3, which provides guidelines for reracking of spent fuel pools.

The spent fuel from three outages is located in old racks towards the south-end of the pool. The licensee indicated that first the new racks will be installed towards the north-end of the pool separating the spent fuel from the load path as much as possible. Mechanical/electrical interlocks will be provided per NUREG-0612 to prevent movement of the overhead crane load blocks over or within 25 feet horizontal of the spent fuel assembly. After installation of new racks on the north side of the pool, spent fuel assemblies will be moved to the new racks using a fuel handling tool and a spent fuel pit bridge crane. The removal of old racks and installation of new racks will then commence towards the south-end of the pool after installation of the required mechanical/ electrical interlocks.

All load handling will follow safe load paths in the fuel handling building and will not pass over any safe shutdown equipment.

The spent fuel shipping cask cannot be carried over the spent fuel pool due to crane travel limitations as discussed in the Byron Safety Evaluation Report, NUREG-0876, Section 9.1.5. Therefore, storage of spent fuel in the new proposed high density storage racks will not affect the staff's previous acceptance regarding a cask drop accident.

As a result of its review, the staff finds that heavy load handling will be performed in accordance with the guidelines of NUREG-0612 and, therefore, the requirements of GDC 61 and 62 are met as they relate to proper load handling to ensure against an unacceptable release of radioactivity or a criticality accident as a result of a heavy load drop.

5.3 Conclusion

Based on the above, the staff concludes that the proposed spent fuel pool storage capacity expansion to accommodate 2870 fuel assemblies with respect to the spent fuel pool cooling system capabilities and handling of heavy loads is acceptable.

6 RADIATION PROTECTION AND ALARA CONSIDERATIONS

To date, only three refueling processes have occurred at this station. Thus, the proposed operations will be conducted in a relatively clean radiological environment. Existing spent fuel consists of approximately one core and occupies one rack in the south portion of the pool. The licensee plans to install new, high density racks in the north section of the pool, move the present spent fuel to the north section, survey, clean (if necessary) and resume reracking.

The proposed plan fundamentally meets the objectives of keeping occupational radiation exposures to a level that is as low as reasonably achievable (ALARA), i.e., reracking operations will occur in pool areas as remote as possible from currently stored spent fuel. Nevertheless, the licensee has committed to the implementation of numerous controls to assure radiation protection objectives are met:

1. Pre-job surveys, daily-verification, and ALARA pre-job meetings will be conducted.

2. The spent fuel pool filter and demineralizer will be operated to reduce radioactivity levels in the pool.

3. An underwater vacuum system will be used for pool cleaning if necessary.

4. Results of radiation surveys will be used to guide divers around "hot spots" in the pool, if any.

5. Underwater communication with divers will be maintained.

6. Divers will be monitored by passive and active dosimeters. At least one of the active dosimeters (probes) will have a high dose rate alarm set-print capability. Either a radiation protection technician or a health physicist will provide timekeeping for the diver(s).

7. Depth of water shielding above the fuel will be maintained at 10 feet; this will assure a dose rate of 2.5 mrem/hr, or less, at the surface of the water from all radiation sources.

8. The station's health physics program includes provisions for air monitoring to detect and control workers' exposures to airborne radionuclides.

The licensee's estimated collective occupational exposure for the reracking is 1.1 person-rem. This is a small fraction of the station's annual collective dose for 1987 which totalled 769 person-rem. The staff finds this estimate to be reasonable and acceptable.

High density storage could increase dose rates external to the pool walls. This can be alleviated readily by placing oldest fuel nearest the sides of the pool. In this regard, the licensee states: "Normal health physics surveys would note any elevation in radiation levels and appropriate action would be taken to assure (that) no personnel hazard exists." The staff finds this appropriate, readily achievable, and acceptable.

7 ACCIDENTS AT HIGHER BURNUP

There will be no significant impacts on any accident dose estimates due to the increase in allowable fuel burnup to 48,000 MWD/T. Accident dose estimates are proportional to power level and enrichment assumptions, predominantly, and those do not change. Burnup to 60,000 MWD/T could increase estimates of accident doses by a factor of about 1.2. For a burnup of 48,000 MWD/T, the increase should be less, but a 1.2 factor is conservatively high.

In its Safety Evaluation Report (NUREG-0876), the NRC staff conservatively estimated doses due to a fuel handling accident. Doses at the exclusion area boundary and low population zone were previously estimated as shown in the following table. Also shown in this table are the doses multiplied by 1.2 account (conservatively) for higher fuel burnup. All these doses are well within the applicable regulatory requirements at 10 CRR Part 100, and are, therefore, acceptable.

RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING DESIGN-BASIS ACCIDENT

	Exclusion area boundary dose, rems		Low population zone dose, rems	
	Thyroid	Whole body	Thyroid	Whole body
Original estimates (NUREG-0876)	29	0.6	1.0	0.1
Estimates for higher fuel burnup*	36	0.7	1.2	0.12
Regulatory requirement (10 CFR Part 100)	300	25	300	25

*Factor of 1.2 greater than original estimate.

8 ENVIRONMENTAL CONSIDERATIONS

The Commission has prepared and published in the Federal Register (54 FR 10758, March 15, 1989) an Environmental Assessment related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement related to the Operation of Byron Station, Units 1 and 2 dated April 1982.

9 CONCLUSIONS

The staff has reviewed and evaluated the licensee's request for the expansion of the spent fuel pool capacity. Based on the considerations discussed in this safety evaluation, the staff concludes that the analyses of the spent fuel rack modules and the spent fuel pool are in compliance with the acceptance criteria set forth in the FSAR and are consistent with the current licensing practice, and therefore, are acceptable.

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