AUGUST 1 2 1983

(2)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: OCONEE UNIT NO. 3 - SPENT FUEL POOL EXPANSION

Purpose: To advise the Commission that the staff is publishing the enclosed notice of consideration and proposed no significant hazards consideration (NSHC) determination relative to the licensee-requested expansion of the Oconee Unit 3 spent fuel pool.

Background: By letter dated March 10, 1983, Duke Power Company (DPC or the licensee) submitted a proposed amendment to the Oconee station operating license and the proposed revision to the Technical Specifications. The proposed Technical Specifications revision would allow the expansion of the Unit 3 spent fuel pool from 474 to 825 spaces by means of reracking the pool with high density neutron absorbing (poison) racks.

The staff reviewed a detailed NSHC determination included in the licensee's submittal and concluded that the determination appears to demonstrate that the three standards specified in 10 CFR 50.92 are met. In this instance, the reracking technology has been well developed and demonstrated in prior rerackings at the Oconee station. The proposed reracking does not appear to create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed reracking would not appear to significantly reduce the margin of safety from the viewpoint of nuclear criticality or thermal-hydraulic, mechanical, material and structural considerations. In view of this, the staff proposes to determine that the licensee's application does not involve a significant hazard consideration.

Contact:
R. Hernan
X-27900

The staff submitted its proposed NSHC determination, as well as the licensee's request, to the Commission on June 23, 1983 (SECY 83-249). Subsequently, the Commission evaluated the staff's proposal. However, the vote on the proposal was split, 2-2. The staff was informed that the General Counsel, on July 27, 1983, advised the Commission that the 2-2 vote permits the staff to proceed with the proposed action or to seek more definitive guidance from the Commission.

Discussion: The staff has elected to proceed with publication in the FEDERAL REGISTER of the notice of consideration of the requested amendment and proposed NSHC determination in order to minimize impacts of further delaying issuance of this proposed amendment. The licensee had planned to commence the reracking operation on or about September 1, 1983 in order to support future refueling outages at the Oconee facility. The licensee, at our request, has provided additional information regarding the impacts of further delaying action on this amendment request. This information is contained in the enclosed letter from DPC dated August 8, 1983.

S/ JACK ROE for

William J. Dircks
Executive Director for Operations

Enclosure:
As Stated

cc w/enclosure:
OGC
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Regional Offices
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4. ~~RCLainas~~ 11/3/83		
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REMARKS

6. ~~D. Eisenhut~~ 11/14

~~T. H. Denton~~

~~W. J. Dircks~~

Not to be sent

Eisenhut — OK to issue w/o going to Commission 11/23

RE: Commission paper - Trojan Nuclear Plant - Spent Fuel Pool Rerack Application

DMS 11/16

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	432
C. M. Trammell	Phone No. 27389

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note Ed Case & HRD's notes —
Go ahead w/FR notice, etc.
NO need to go to Comm.
11/23

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~~2. CTrammell~~ CMT 9/20/83		
~~3. JRMiller~~ JRM 9/26/83		
~~4. GCLainas~~		
5. JGray - OELD		

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Coordination	Justify	

REMARKS

6. D. Eisenhut
7. H. Denton
8. W. J. Dircks

RE: Commission paper - Trojan Nuclear Plant - Spent Fuel Pool Rerack Application

Note that our review has started & is scheduled for completion by Jan. 1984. The licensee plans to start fabrication only after approval.

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

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C. M. Trammell	Phone No. 27389

5041-102

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REMARKS

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For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: TROJAN NUCLEAR PLANT - SPENT FUEL POOL RERACK APPLICATION

Purpose: To inform the Commission that the staff is publishing the enclosed Federal Register Notice which contains a proposed determination that the Trojan rerack application for the spent fuel pool does not involve a significant hazards consideration.

Discussion: By letter dated August 1, 1983, as amended October 31, 1983, Portland General Electric Company submitted a proposed amendment to the Trojan operating license which would authorize the licensee to increase the storage capacity of the spent fuel pool from the present capacity of 651 fuel assemblies to 1408 fuel assemblies (second rerack). The change would be accomplished by reracking the pool with high density neutron absorbing racks. A copy of the licensee's submittal is also enclosed.

When the Commission approved the Interim Final Rule "Standards for Determining Whether License Amendments Involve No Significant Hazards Considerations", spent fuel pool reracking was specifically excluded from the list of examples considered likely to involve a significant hazards consideration. The Commission stated that it would be making a finding on the question of no significant hazards consideration for each reracking application (such as this) on a case-by-case basis, giving full consideration to the technical circumstances of the case, using the standards of §50.92 (48 FR 14869).

The staff has reviewed the detailed no significant hazards consideration determination included in the licensee's submittal (Attachment 1, pp. 3-10 of the enclosed application) and has concluded that the determination appears to demonstrate that the three standards of §50.92 are satisfied.

Contact:
C. Trammell, NRR
49-27389

A similar notice was issued on August 16, 1983 (48 FR 37108) with respect to the Oconee Unit 3 rerack application.

The proposed no significant hazards consideration determination for the Trojan rerack is also consistent with the conclusion of the staff's Information Report "Study on Significant Hazards" (SECY-83-337, August 15, 1983). This study concluded that a request to expand the storage capacity of a spent fuel pool which satisfies the following is considered not likely to involve a significant hazards consideration:

(1) The storage expansion method consists of either replacing existing racks with a design which allows closer spacing between stored spent fuel assemblies or placing additional racks of the original design on the pool floor if space permits,

(2) The storage expansion method does not involve rod consolidation or double-tiering,

(3) The k_{eff} of the pool is maintained less than or equal to 0.95, and

(4) No new technology or unproven technology is utilized in either the construction process or the analytical techniques necessary to justify the expansion.

The Trojan application appears to meet these criteria as well as the three standards of §50.92.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Federal Register Notice
2. PGE Rerack Application dated 8-1-83

A similar notice was issued on August 16, 1983 (48 FR 37108) with respect to the Oconee Unit 3 rerack application.

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(4) No new technology or unproven technology is utilized in either the construction process or the analytical techniques necessary to justify the expansion.

The Trojan application appears to meet these criteria as well as the three standards of §50.92.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Federal Register Notice
2. PGE Rerack Application dated 8-1-83

				D:NRR	EDO
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UNITED STATES NUCLEAR REGULATORY COMMISSION

PORTLAND GENERAL ELECTRIC COMPANY, ET AL.

DOCKET NO. 50-344

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

The U. S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF-1, issued to Portland General Electric Company, Pacific Power and Light Company, and The City of Eugene, Oregon (the licensee), for operation of the Trojan Nuclear Plant located in Columbia County, Oregon.

The amendment would authorize the licensee to increase the storage capacity of the spent fuel pool from the present capacity of 651 fuel assemblies to 1408 fuel assemblies. The change would be accomplished by the installation of spent fuel racks having a closer spacing and a modified nuclear design. The present racks have a cell spacing of 13.3 inches. Under the proposed amendment, the cell spacing would be reduced to 10.5 inches and the racks would utilize neutron absorbing material between cells to assure a sub-critical configuration. Also, the amendment would increase the authorized enrichment of fuel in the pool from the present 3.5% U-235 to 4.5% U-235 to accommodate possible use and storage of fuel of this higher enrichment at a later time. To provide more room for storage racks, the licensee also proposes to remove the spent fuel pool cooling sparger line which currently forms a ring inside the perimeter of the spent fuel pool floor. Finally, the amendment would prohibit the licensee from moving any spent fuel shipping casks into the building containing the spent fuel pool. (Removal of this restriction would require NRC review and approval at a later time.) The amendment request is provided in a letter dated August 1, 1983, and Amendment 1 dated October 31, 1983, together with a technical report designated as PGE-1037, "Trojan Nuclear Plant Spent Fuel Storage Rack Replacement Report."

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility or a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983 (48 FR 14870). Spent fuel pool reracking was specifically excluded from the list of examples considered likely to involve a significant hazards consideration. Pending further study of this matter, the Commission is making a finding on the question of no significant hazards consideration for each reracking application such as this on a case-by-case basis, giving full consideration to the technical circumstances of the case, using the standards of §50.92 (48 FR 14869).

The licensee's submittal of August 1, 1983 and Amendment 1 of October 31, 1983 included a discussion of the proposed action with respect to the issue of no significant hazards consideration. This discussion has been reviewed and the Commission finds it acceptable. Pertinent portions of the licensee's discussion of this matter, addressing each of the three standards, is presented below.

> In general consideration, this amendment does little more than allow the storage of spent fuel assemblies that have greater than nine years' decay after discharge to the SFP. The additional 757 assemblies that could be stored will have a much lower heat generation rate and radioactivity content than the 651 assemblies currently allowed to be stored, and, therefore will increase the total SFP heat load and radioactivity content by only a small amount. The storage of recently discharged spent fuel has already been approved by the NRC.
>
> The replacement spent fuel storage racks are of the freestanding, neutron absorber type of design without attachments to each other or the SFP (sliding is permitted under lateral loading). Racks of this type designed by Nuclear Energy Services, Inc. (the vendor for the Trojan racks) have been licensed for use at five nuclear plants, and racks of similar design by other vendors are in use at many nuclear plants.
>
> <u>First Standard</u>
>
> Involve a significant increase in the probability or consequences of an accident previously evaluated.
>
> Analysis of this proposed spent fuel rack replacement has been accomplished using current NRC Staff accepted Codes and Standards as specified in Chapter 2 of PGE-1037. The results of the analysis show that the specified acceptance criteria set forth in these standards are met.
>
> Probabilities
>
> The following potential accident scenarios have been identified and are discussed in Chapters 3 and 4 of PGE-1037:
>
> 1) Seismic events.

2) Tornado-generated missile impacts.

3) Load drops, including a fuel handling accident.

4) Loss of SFP forced cooling.

5) Criticality accidents.

6) Installation accidents.

The probability of an occurrence for any of the first four accidents is not affected by the racks themselves, since they are essentially initiating events; thus, rack replacement cannot increase the probability of these accidents.

The probability of a criticality accident is discussed in Section 3.1 of PGE-1037. The racks were evaluated against the guidelines, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications". All potential events that could involve accidental criticality were examined. It was concluded that the only event that could result in accidental criticality was the placement of an assembly adjacent to a loaded storage rack during rack replacement. This will be precluded by administrative controls during rack replacement requiring a vacant row of cells be maintained along the exposed side of the racks containing fuel. Therefore, the probability of a criticality accident will not be increased over that which was evaluated by the NRC in their review of the previous Trojan rack replacement submittal (License Amendment 34, November 3, 1978).

In regard to installation accidents, Sections 3.3.3 and 5.1 of PGE-1037 describe the analysis of installation accidents. As indicated in these sections, precautions acceptable to the NRC Staff will be taken via procedures and interlocks on the SFP bridge crane to preclude the movement of racks or other "heavy" loads over spent fuel. Thus, the proposed Trojan SFP rack replacement will not involve an increase in probability of an accident over that which was evaluated by the NRC in their review of the previous Trojan rack replacement submittal.

Consequences

The consequences of a design basis seismic event have been evaluated and are described in Section 3.3.3 of PGE-1037. The racks were evaluated against the appropriate standards described in Section 2.3 of PGE-1037. The results of the analysis show that the proposed racks meet all of the NRC structural acceptance criteria applicable to Trojan, and are consistent with results found acceptable by the NRC Staff in the previous Trojan rack replacement Safety Evaluation Report (November 11, 1977). Thus, the consequences of seismic events for the new storage racks will not significantly increase from those previously evaluated for the present storage racks.

The consequences of tornado missile impacts have been analyzed and are described in Section 3.3.3 of PGE-1037. The racks were evaluated against Trojan design basis tornado missiles and the appropriate standards as described in Section 2.3 of PGE-1037. The results of this analysis show that the accident consequences will not exceed those postulated for the fuel handling accident described in the Trojan Updated FSAR, Section 15.7.4 [The analysis and consequences in the Updated FSAR are unchanged from that in the original FSAR, which was reviewed and accepted by the NRC, and documented as such in the Trojan Safety Evaluation Report.] Thus, the consequences of tornado missile impacts will not increase from previously evaluated events.

Load drop accidents potentially include both "light" loads, which have an impact energy less than the limit specified in the Trojan Technical Specifications (240,000 in.-lbs), and "heavy" loads, as described in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". The consequences of load drop accidents other than "heavy" loads have been evaluated and are described in Section 3.3.3 of PGE-1037. The racks were evaluated in accordance with the appropriate criteria as described in Section 2.3. The results of this analysis show that the accident consequences will not exceed those postulated for the fuel handling accident described in the Trojan Updated FSAR, Section 15.7.4 [The analysis and consequences in the Updated FSAR are unchanged from that in the original FSAR, which was reviewed and accepted by the NRC, and documented as such in the Trojan Safety Evaluation Report]. Thus, the consequences of "light" load drop accidents will not increase from previously evaluated accidents.

Section 4.2.5 of PGE-1037 discusses "heavy" load drop accidents. As explained in Section 4.2.5, with the possible exception of a spent fuel shipping cask, no "heavy" load drops into the SFP are credible. In regard to the spent fuel assembly shipping cask, Amendment 1 to LCA 94 includes a change to Page 5g of License NPF-1 which prohibits the movement of a spent fuel assembly shipping cask into the Fuel Building. Therefore, the consequences of "heavy" load drops will not increase from previously evaluated accidents.

The consequences of a loss of SFP forced cooling have been evaluated and are described in Section 3.2.2 of PGE-1037. As indicated in Section 3.2.2, if a loss of SFP forced cooling should occur, there is ample time to effect repairs to the cooling system or to establish a makeup flow. The maximum water boiloff rate of 95-gpm is less than the 200-gpm makeup rate given in the Trojan Updated FSAR, Section 9.1.3 [The analysis and consequences in the Updated FSAR are unchanged from that in the original FSAR, which was reviewed and accepted by the NRC, and documented as such in the Trojan Safety Evaluation Report.] Therefore, the consequences of this type of accident will not be significantly increased from previously evaluated accidents by this proposed rack replacement.

The consequences of a criticality accident are analyzed in Section 3.1 of PGE-1037. As indicated above, it has been determined that, with the inclusion of administrative controls to maintain a vacant row of cells along the exposed side of the racks containing fuel during rack installation, there are no postulated events which will result in a criticality accident. Therefore, the consequences of a criticality accident are not increased from the consequences previously evaluated by the NRC for the prior rack replacement.

The consequences of an installation accident (ie, dropping of a spent fuel rack or other "heavy" load during rack replacement) are analyzed in Sections 3.3.3 and 5.1 of PGE-1037. The consequences were evaluated against the criteria described in Section 2.3. As indicated in Sections 3.3.3 and 5.1, precautions will be taken via administrative procedures and interlocks on the SFP bridge crane to preclude the movement of racks or other "heavy" loads over spent fuel. Thus, the consequences of an accident during rack replacement will not be significantly increased from previously evaluated accidents.

Therefore, it is shown that the proposed Trojan spent fuel rack replacement will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

- Create the possibility of a new or different kind of accident from any accident previously evaluated.

PGE has evaluated the proposed rack replacement in accordance with the "NRC Position for Review and Acceptance of Spent Fuel

Storage and Handling Applications", appropriate NRC Regulatory Guides, appropriate NRC Standard Review Plan sections, and appropriate industry Codes and Standards as described in Chapter 2 of PGE-1037. In addition, PGE has reviewed the NRC Safety Evaluation Report for the previous Trojan spent fuel rack replacement application.

The conclusion of this review is that the proposed rack replacement does not create the possibility of a new or different kind of accident from any previously evaluated. All possible accidents have been previously analyzed and evaluated for the original spent fuel storage racks and the prior rack replacement. As discussed in the previous section, a cask drop accident cannot occur since no casks will be moved into the Fuel Building at Trojan.

Third Standard

Involve a significant reduction in a margin of safety.

The issue of margin of safety when applied to a spent fuel rack replacement modification needs to address the following areas (as established by the NRC Staff Safety Evaluation review process):

a. Nuclear criticality considerations.

b. Thermal hydraulic considerations.

c. Mechanical, material, and structural considerations.

The margin of safety that has been established for nuclear criticality considerations is that the neutron multiplication factor in the SFP is to be $\leq$ to 0.95, including all uncertainties, under all conditions. For the proposed modification, the criticality analysis is described in Section 3.1 of PGE-1037.

The methods utilized in the analysis conform with ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations"; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"; and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications". The computer programs, data libraries, and benchmarking data used in the evaluation have been used in previous spent fuel rack replacement applications by other NRC licensees and have been reviewed and approved by the NRC. The results of this analysis indicate that k_{eff} is < 0.95 under all postulated conditions, including uncertainties at a 95/95 probability/confidence level. Thus, meeting the acceptance criteria for criticality, the proposed rack replacement does not involve a significant reduction in the margin of safety for nuclear criticality.

From a thermal hydraulic consideration, the areas of concern when evaluating if there is a significant reduction in margin of safety are: (1) maximum fuel temperature, and (2) the increase in temperature of the water in the pool. The thermal hydraulic evaluation is described in Section 3.2 of PGE-1037. Results of these analyses show that fuel cladding temperatures under abnormal conditions are sufficiently low to preclude structural failure and that boiling does not occur in the water channels between the fuel assemblies nor within the storage cells. However, the proposed rack replacement will result in an increase in the maximum heat load in the Trojan SFP. As shown in Section 3.2, the maximum SFP temperature will not exceed the current margin of safety (140°F) given in Trojan Updated FSAR Section 9.1.3 for a normal refueling. For the maximum normal heat load case (full-core discharge at 150 hr after shutdown, which fills the SFP to its capacity), the SFP temperature will not exceed 140°F unless the temperature of the Columbia River rises above 69°F. Under extreme Columbia River water temperatures the maximum calculated SFP temperature is 146°F, which will fall below 140°F after an additional 33 hr of spent fuel decay time. This maximum temperature increase above 140°F for 33 hr is not significant from a safety standpoint. In addition, since SFP water temperature is continuously monitored and alarmed in the control room, appropriate actions can be taken should the SFP water temperature approach 140°F during refueling operations. Thus, it is concluded that the margin of safety of 140°F described in Trojan Updated FSAR Section 9.1.3 will not be significantly reduced by this SFP rack replacement.

The mechanical, material, and structural considerations of the proposed rack replacement are analyzed in Section 3.3 of PGE-1037. As described in Section 3.3.3, the racks are designed in accordance with the applicable NRC Regulatory Guides, Standard Review Plan sections, and position papers, as well as the appropriate industry Codes and Standards. The racks are designed to Seismic Category I requirements and are classified as ASME Code Class 3 Component Support Structures. The materials utilized are described in Section 3.3 and are compatible with the SFP and the spent fuel assemblies. The conclusion of the analysis in Section 3.3 is therefore that the margin of safety is not significantly reduced by the proposed rack replacement.

Thus, it has been shown that the proposed Trojan SFP rack replacement does not:

a. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

b. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

c. Involve a significant reduction in a margin of safety.

Because the submittal and above discussion presented by the licensee appear to demonstrate that the standards specified in 10 CFR 50.92 are met, and because the reracking technology in this instance has been well developed and demonstrated, the Commission proposes to determine that operation of the facility in accordance with the proposed amendment does not involve a significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Comments should be addressed to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Docketing and Service Branch.

By , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to James R. Miller: petitioner's name and telephone number;

date petition was mailed; plant name; and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to J. W. Durham, Senior Vice President, Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the local public document room located at the Multnomah County Library, Social Science and Science Department, 801 S.W. 10th Avenue, Portland, Oregon 97205.

Dated at Bethesda, Maryland, this

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

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

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FEB 2 1 1984

Docket No. 50-344

Mr. Bart D. Withers
Vice President Nuclear
Portland General Electric Company
121 S. W. Salmon Street
Portland, Oregon 97204

Dear Mr. Withers:

By your letters dated November 23 and December 30, 1983 and affidavits dated November 16 and December 30, 1983, you submitted Trojan Nuclear Plant spent fuel storage rack design drawings and design calculations prepared by Nuclear Energy Services, Inc. (NES) and requested that they be withheld from public disclosure pursuant to 10 CFR 2.790.

NES stated that the submitted information should be treated as proprietary for the following reasons:

1. It is an NES policy to maintain the confidentiality of design drawings due to the detailed information contained therein. In the case of spent fuel rack designs, it is essential that the drawings be prevented from entering the public domain because spent fuel rack contracts are competitively awarded. If the detailed characteristics of NES' design and fabrication processes were made public, it would adversely affect NES' competitive position within the industry.

2. It is an NES policy to maintain the confidentiality of design documents due to the detailed design information and analysis techniques contained therein. In the case of spent fuel designs, it is essential that the documents be prevented from entering the public domain because spent fuel rack contracts are competitively awarded. If the detailed characteristics of NES' design and analysis techniques were made public, it would adversely affect NES' competitive position within the industry.

3. NES is consistent in the application of this policy regarding design documents for fuel rack projects.

We have reviewed your application and the material based on the requirements and criteria of 10 CFR 2.790 and, on the basis of NES' statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information.

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It is our belief, pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended, that, at this time, the right of the public to be fully apprised of the submitted information does not outweigh the need to protect NES' competitive position.

Accordingly, we have determined that the information should be withheld from public disclosure.

We therefore, approve your request for withholding pursuant to 10 CFR 2.790 and are withholding the following documents from public inspection as proprietary:

NES DRAWING NO.

80B7696	80C7690	80D7698	80E7684
80B7697	80C7691	80D7699	80E7685
80B7735	80C7692	80D7700	80E7686
80B7736	80C7693	80D7702	80E7687
80B7753	80C7694	80D7703	80E7688
80B7754	80C7695	80D7704	80E7689
	80C7701	80D7705	80E7740
	80C7708	80D7706	80E7743
	80C7709	80D7707	80E7744
	80C7710	80D7748	80E7745
	80C7711	80D7750	80E7746
	80C7712	80D7751	80E7747
	80C7713	80D7752	
	80C7737		
	80C7749		

Twelve pages of weld stress calculations prepared by NES (identified as "prepared by J. Shah, Project 5529, 5240, Task 320" dated December 13 1983 (pages 1, 2, 3, 4, 10) and December 9, 1983 (pages 5, 6, 7, 8, 9, 11, 12) entitled "NRC Licensing Support" and numbered pages 1 through 12.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the documents. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, insure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You should also understand that the NRC may have cause to review this determination in the future, such as if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC needs additional information from you or makes a determination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

cc: See next page

Portland General Electric Company
(cc list for Spent Fuel Pool Proceeding only)

cc: Ivan W. Smith, Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Peter A. Morris
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Oscar H. Paris
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Robert M. Hunt, Chairman
Board of County Commissioners
Columbia County
St. Helens, Oregon 97501

Regional Administrator
Nuclear Regulatory Commission, Region V
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Walter Perry, III
Attorney for Oregon Deparment
of Energy and Energy Facility
Siting Council
100 Justice Building
Salem, OR 97310

Ronald Johnson, Esq.
Portland General Electric Co.
121 S.W. Salmon Street
Portland, OR 97204

Eugene Rosolie
Coalition for Safe Power
408 S.W. Second Street
Suite 410
Portland, OR 97204

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

Circulate
(5)

January 18, 1979

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US NRC
NRR DOR ENGINEERING & PROJECTS
ASSISTANT DIRECTOR
542
WASHINGTON DC 20555

To All Power Reactor Licensees

Gentlemen:

Our letter of April 14, 1978, provided NRC Guidance entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications." Enclosed are modifications to this document for your information and use. These involve pages IV-5 and IV-6 of the document and comprise modified rationale and corrections.

Sincerely,

Brian K. Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosure:
Pages IV-5 and IV-6

cc w/enclosure:
Service List

~~7903080100~~

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described in "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below. When buckling loads are considered in the design, the structural acceptance criteria shall be limited by the requirements of Appendix XVII-2110(b) of the ASME Boiler and Pressure Vessel Code.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	Acceptance Limit
Elastic Analysis	
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	Lesser of 2Sy or Su stress range
D + L + To + E	Lesser of 2Sy or Su stress range
D + L + Ta + E	Lesser of 2Sy or Su stress range
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.
2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.
3. The provisions of NF 3231.1 shall be amended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

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

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorizatic to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

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Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

OT POSITION FOR REVIEW AND ACCEPTANCE OF SPENT FUEL STORAGE AND HANDLING APPLICATIONS

I. BACKGROUND

Prior to 1975, low density spent fuel storage racks were designed with a large pitch, to prevent fuel pool criticality even if the pool contained the highest enrichment uranium in the light water reactor fuel assemblies. Due to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better utilize available space. In the case of operating plants the new rack system interfaces with the old fuel pool structure. A proposal for installation of high density storage racks may involve a plant in the licensing stage or an operating plant. The requirements of this position do not apply to spent fuel storage and handling facilities away from the nuclear reactor complex.

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, an environmental impact statement or environmental impact appraisal shall be prepared in which five specific factors in addition to the normal cost/benefit balance and environmental stresses should be applied, balanced and weighed.

The overall design objectives of a fuel storage facility at the reactor complex are governed by various Regulatory Guides, the Standard Review Plan, and industry standards which are listed in the reference section. Based on the reviews of such applications to date it is obvious that the staff had to request additional information that could be easily included in an adequately documented initial submittal. It is the intent of this document to provide guidance for the type and extent of information needed to perform the review, and to indicate the acceptance criteria where applicable.

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II. REVIEW DISCIPLINES

The objective of the staff review is to prepare (1) Safety Evaluation Report, and (2) Environmental Impact Appraisal. The broad staff disciplines involved are nuclear, mechanical, material, structural, and environmental.

Nuclear and thermal-hydraulic aspects of the review include the potential for inadvertant criticality in the normal storage and handling of the spent fuel, and the consequences of credible accidents with respect to criticality and the ability of the heat removal system to maintain sufficient cooling.

Mechanical, material and structural aspects of the review concern the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand the effects of natural phenomena such as earthquakes, tornadoes, flood, effects of external and internal missiles, thermal loading, and also other service loading conditions.

The environmental aspects of the review concern the increased thermal and radiological releases from the facility under normal as well as accident conditions, the occupational radiation exposures, the generation of radioactive waste, the need for expansion, the commitment of material and nonmaterial resources, realistic accidents, alternatives to the proposed action and the cost-benefit balance.

The information related to nuclear and thermal-hydraulic type of analyses is discussed in Section III.

The mechanical, material, and structural related aspects of information are discussed in Section IV.

The information required to complete an environmental impact assessment, including the five factors specified by the Commission, is provided in Section V.

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

CPE

1. Neutron Multiplication Factor

To include all credible conditions, the licensee shall calculate the effective neutron multiplication factor, k_{eff}, in the fuel storage pool under the following sets of assumed conditions:

1.1 Normal Storage

a. The racks shall be designed to contain the most reactive fuel authorized to be stored in the facility without any control rods or any noncontained* burnable poison and the fuel shall be assumed to be at the most reactive point in its life.

b. The moderator shall be assumed to be pure water at the temperature within the fuel pool limits which yields the largest reactivity.

c. The array shall be assumed to be infinite in lateral extent or to be surrounded by an infinitely thick water reflector and thick concrete,** as appropriate to the design.

d. Mechanical uncertainties may be treated by assuming "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

e. Credit may be taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption, provided a means of inspection is established (refer to Section 1.5).

1.2 Postulated Accidents

The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies. The

*"Noncontained" burnable poison is that which is not an integral part of the fuel assembly.

**It should be noted that under certain conditions concrete may be a more effective reflector than water.

postulated accidents shall include: (1) dropping of a fuel element on top of the racks and any other achievable abnormal location of a fuel assembly in the pool; (2) a dropping or tipping of the fuel cask or other heavy objects into the fuel pool; (3) effect of tornado or earthquake on the deformation and relative position of the fuel racks; and (4) loss of all cooling systems or flow under the accident conditions, unless the cooling system is single failure proof.

1.3 Calculation Methods

The calculation method and cross-section values shall be verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. Sufficiently diverse configurations shall be calculated to render improbable the "cancellation of error" in the calculations. So far as practicable the ability to correctly account for heterogeneities (e.g., thin slabs of absorber between storage locations) shall be demonstrated.

A calculational bias, including the effect of wide spacing between assemblies shall be determined from the comparison between calculation and experiment. A calculation uncertainity shall be determined such that the true multiplication factor will be less than the calculated value with a 95 percent probability at a 95 percent confidence level. The total uncertainity factor on k_{eff} shall be obtained by a statistical combination of the calculational and mechanical uncertainties. The k_{eff} value for the racks shall be obtained by summing the calculated value, the calculational bias, and the total uncertainty.

1.4 Rack Modification

For modification to existing racks in operating reactors, the following information should be provided in order to expedite the review:

(a) The overall size of the fuel assembly which is to be stored in the racks and the fraction of the total cell area which represents the overall fuel assembly in the model of the nominal storage lattice cell;

(b) For H_2O + stainless steel flux trap lattices; the nominal thickness and type of stainless steel used in the storage racks and the thermal (.025 ev) macroscopic neutron absorption cross section that is used in the calculation method for this stainless steel;

(c) Also, for the H_2O + stainless steel flux trap lattices, the change of the calculated neutron multiplication factor of infinitely long fuel assemblies in infinitely large arrays in the storage rack (i.e., the k_∞ of the nominal fuel storag lattice cell and the changed k_∞) for:

(1) A change in fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly where it assumed that this change is made by increasing the enrichment of the U^{235}; and,

(2) A change in the thickness of stainless steel in the storage racks assuming that a decrease in stainless steel thickness is taken up by an increase in water thickness and vice versa;

(d) For lattices which use boron or other strong neutron absc ers provide:

(1) The effective areal density of the boron-ten atoms (i.e., B^{10} atoms/cm^2 or the equivalent number of bo ten atoms for other neutron absorbers) between fuel assemblies.

(2) Similar to Item C, above, provide the sensitivity c the storage lattice cell k_∞ to:

(a) The fuel loading in grams of U^{235}, or equival per axial centimeter of fuel assembly,

(b) The storage lattice pitch; and,

(c) The areal density of the boron-ten atoms bet fuel assemblies.

1.5 Acceptance Criteria for Criticality

The neutron multiplication factor in spent fuel pools shall less than or equal to 0.95, <u>including all uncertainties</u>, ur all conditions

(1) For those facilities which employ a strong neutron ab material to reduce the neutron multiplication factor storage pool, the licensee shall provide the descript onsite tests which will be performed to confirm the p and retention of the strong absorber in the racks. results of an initial, onsite verification test shal within 95 percent confidence limits that there is a -cient amount of neutron absorber in the racks to mai the neutron multiplication factor at or below 0.95. addition, coupon or other type of surveillance test be performed on a statistically acceptable sample s

periodic basis throughout the life of the racks to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95.

(2) Decay Heat Calculations for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 75/087).

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

Conservative methods should be used to calculate the maximum fuel temperature and the increase in temperature of the water in the pool. The maximum void fraction in the fuel assembly and between fuel assemblies should also be calculated.

Ordinarily, in order not to exceed the design heat load for the spent fuel cooling system it will be necessary to do a certain amount of cooling in the reactor vessel after reactor shutdown prior to moving fuel assemblies into the spent fuel pool. The bases for the analyses should include the established cooling times for both the usual refueling case and the full core off load case.

A potential for a large increase in the reactivity in an H_2O flux trap storage lattice exists if, somehow, the water is kept out or forced out of the space between the fuel assemblies, conceivably by trapped air or steam. For this reason, it is necessary to show that the design of the storage rack is such that this will not occur and that these spaces will always have water in them. Also, in some cases, direct gamma heating of the fuel storage cell walls and of the intercell water may be significant. It is necessary to consider direct gamma heating of the fuel storage cell walls and of the intercell water to show that boiling will not occur in the water channels between the fuel assemblies. Under postulated accident conditions where all non-Category I spent fuel pool cooling systems become inoperative, it is necessary to show that there is an alternate method for cooling the spent pool water. When this alternative method requires the installation of alternate components or significant physical alteration of the cooling system, the detailed steps shall be described, along with the time required for each. Also, the average amount of water in the fuel pool and the expected heat up rate of this water assuming loss of all cooling systems shall be specified.

(4) Potential Fuel and Rack Handling Accidents

The method for moving the racks to and from and into and out of the fuel pool, should be described. Also, for plants where the spent fuel pool modification requires different fuel handling procedures than that described in the Final Safety Analysis Report, the differences should be discussed. If potential fuel and rack handling accidents occur, the neutron multiplication factor in the fuel pool shall not exceed 0.95. These postulated accidents shall not be the cause of the loss of cooling for either the spent fuel or the reactor.

(5) Technical Specifications

To insure against criticality, the following technical specifications are needed on fuel storage in high density racks:

1. The neutron multiplication factor in the fuel pool shall be less than or equal to 0.95 at all times.

2. The fuel loading (i.e., grams of uranium-235, or equivalent, per axial centimeter of assembly) in fuel assemblies that are to be loaded into the high density racks should be limited. The number of grams of uranium-235, or equivalent, put in the plant's technical specifications shall preclude criticality in the fuel pool.

Excessive pool water temperatures may lead to excessive loss of water due to evaporation and/or cause fogging. Analyses of thermal load should consider loss of all pool cooling systems. To avoid exceeding the specified spent fuel pool temperatures, consideration shall be given to incorporating a technical specification limit on the pool water temperature that would resolve the concerns described above. For limiting values of pool water temperatures refer to ANSI-N210-1976 entitled, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," except that the requirements of the Section 9.1.3.III.1.d of the Standard Review Plan is applicable for the maximum heat load with normal cooling systems in operation.

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool and the racks. The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed under the environmental discipline. Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. Integrity of the racks and the fuel pool due to a postulated fuel handling accident is reviewed under the mechanical, material, and structural disciplines. Sketches and sufficient details of the fuel handling system should be provide to facilitate this review.

(2) Applicable Codes, Standards and Specifications

Construction materials should conform to Section III, Subsection NF of the ASME* Code. All Materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for stainless steel base metal may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code.

Other materials, design procedures, and fabrication techniques will be reviewed on a case by case basis.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61 respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guide 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.

**American Institute of Steel Construction, Latest Edition.

The peak response from each direction should be combined by square root of the sum of the squares. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

The effect of submergence of the rack system on the damping and the mass of the fuel racks has been under study by the NRC. Submergence in water may introduce damping from two sources, (a) viscous drag, and (b) radiation of energy away from the submerged body in those cases where the confining boundaries are far enough away to prevent reflection of waves at the boundaries. Viscous damping is generally negligible. Based upon the findings of this current study for a typical high density rack configuration, wave reflections occur at the boundaries so that no additional damping should be taken into account.

A report on the NRC study is to be published shortly under the title "Effective Mass and Damping of Submerged Structures (UCRL-52342)," by R. G. Dong. The recommendations provided in this report on the added mass effect provide an acceptable basis for the staff review. Increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described in the report: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target materi[al] utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to th[e] applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and bas[e] slab. Temperature gradient across the rack structure ... differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of Section 3.8.4-II.3 of the Standard Review Plan.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spect[ra] corresponding to the highest support elevation provided tha[t] there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and

(b) A static analysis of the rack system by subjecting it to th[e] maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below.

??

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	Acceptance Limit
Elastic Analysis	
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	~~1.5 times normal limits~~, or the lesser of 2 Sy ~~and~~ or Su
D + L + To + E	~~1.5 times normal limits~~, or the lesser of 2 Sy ~~and~~ or Su
D + L + Ta + E	~~1.6 times normal limits~~, or the lesser of 2 Sy or Su
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be ammended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

the construction phase should be provided. Methods for str tural qualification of special poison materials utilized tc absorb neutron radiation should be described. The material the fuel rack is reviewed for compatibility inside the fuel environment. The quality of the fuel pool water in terms c pH value and the available chlorides, fluorides, boron, hea metals should be indicated so that the long-term integrity rack structure, fuel assembly, and the pool liner can be ev

Acceptance criteria for special materials such as poison m should be based upon the results of the qualification prog supported by test data and/or analytical procedures.

If connections between the rack and the pool liner are mad welding, the welder as well as the welding procedure for t welding assembly shall be qualified in accordance with the cable code.

If precipitation hardened stainless steel material is usec the construction of the spent fuel pool racks, hardness te should be performed on each rack component of the subject to verify that each part is heat treated properly. In ad the surface film resulting from the heat treatment should removed from each piece to assure adequate corrosion resi

(8) Testing and Inservice Surveillance

Methods for verification of long-term material stability mechanical integrity of special poison material utilized neutron absorption should include actual tests.

Inservice surveillance requirements for the fuel racks ar poison material, if applicable, are dependent on specifi features. These features will be reviewed on a case by basis to determine the type and the extent of inservice lance necessary to assure long-term safety and integrity pool and the fuel rack system.

V. COST/BENEFIT ASSESSMENT

1. Following is a list of information needed for the environmental Cost/Benefit Assessment:

1.1 What are the specific needs that require increased storage capacity in the spent fuel pool (SFP)? Include in the response:

(a) status of contractual arrangements, if any, with fuel-storage or fuel-reprocessing facilities,

(b) proposed refueling schedule, including the expected number of fuel assemblies that will be transferred into the SFP at each refueling until the total existing capacity is reached,

(c) number of spent fuel assemblies presently stored in the SFP,

(d) control rod assemblies or other components stored in the SFP, and

(e) the additional time period that spent fuel assemblies would be stored onsite as a result of the proposed expansion, and

(f) the estimated date that the SFP will be filled with the proposed increase in storage capacity.

1.2 Discuss the total construction associated with the proposed modification, including engineering, capital costs (direct and indirect) and allowances for funds used during construction.

1.3 Discuss the alternative to increasing the storage capacity of the SFP. The alternatives considered should include:

(a) shipment to a fuel reprocessing facility (if available),

(b) shipment to an independent spent fuel storage facility,

(c) shipment to another reactor site,

(d) shutting down the reactor.

The discussion of options (a), (b) and (c) should include a cost comparison in terms of dollars per KgU stored or cost per assembly The discussion of (d) should include the cost for providing replacement power either from within or outside the licensee's generating system.

1.4 Discuss whether the commitment of material resources (e.g., stainless steel, boral, B_4C, etc.) would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. Describe the material resources that would be consumed by the proposed modification.

1.5 Discuss the additional heat load and the anticipated maximum temperature of water in the SFP which would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems and whether there will be any significant increase in the amount of heat released to the environment.

V.2. RADIOLOGICAL EVALUATION

2. Following is a list of information needed for radiological evaluation:

2.1 The present annual quantity of solid radioactive wastes generated by the SFP purification system. Discuss the expected increase in solid wastes which will result from the expansion of the capacity of the SFP.

2.2. Data regarding krypton-85 measured from the fuel building ventilation system by year for the last two years. If data are not available from the fuel building ventilation system, provide this data for the ventilation release which includes this system.

2.3 The increases in the doses to personnel from radionuclide concentrations in the SFP due to the expansion of the capacity of the SFP, including the following:

(a) Provide a table showing the most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations.

(b) The models used to determine the external dose equivalent rate from these radionuclides. Consider the dose equivalent rate at some distance above the center and edge of the pool respectively. (Use relevant experience if necessary).

(c) A table of recent analysis performed to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

(d) The model and assumptions used to determine the increase, if any, in dose rate from the radionuclides identified in (c) above in the SFP area and at the site boundary.

(e) An estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter media.

(f) The buildup of crud (e.g., ^{58}Co, ^{60}Co) along the sides of the pool and the removal methods that will be used to reduce radiation levels at the pool edge to as low as reasonably achievable.

(g) The expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (e) and (f) above.

A discussion of the radiation protection program as it affects (a) through (g) should be provided.

2.4 Indicate the weight of the present spent fuel racks that will be removed from the SFP due to the modification and discuss what will be done with these racks.

V.3 ACCIDENT EVALUATION 4EB

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

(b) evaluation of the overhead handling system with respect to Regulatory Guide 1.104.

3.2 If the accident aspects of review do not establish acceptability with respect to either (a) or (b) above, then technical specifications may be required that prohibit cask movement in the spent fuel building.

3.3 If the accident review does not establish acceptability with respect to (b) above, then technical specifications may be required that:

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

(2) indicate the minimum age of fuel in pool sections during movement of heavy loads near the pool. In special cases evaluation of consequences-limiting engineered safety features such as isolation systems and filter systems may be required.

...drop/tip analysis as in 3.1(a) above is promis... future submittal, the staff evaluation will include a conc... on the feasibility of a specification of minimum age of fu... based on previous evaluations.

3.5 The maximum weight of loads which may be transported over s... fuel may not be substantially in excess of that of a single assembly. A technical specification will be required to th... effect.

3.6 Conclusions that determination of previous Safety Evaluation Reports and Final Environmental Statements have not changed significantly or impacts are not significant are made so tha... negative declaration with an Environmental Impact Appraisal (rather than a Draft and Final Environmental Statement) can b... issued. This will involve checking realistic as well as conservative accident analyses.

VI. REFERENCES

1. Regulatory Guides

   1.13 - Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

   1.29 - Seismic Design Classification

   1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

   1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

   1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

   1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section III, Division 1

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

   4. American Society of Civil Engineers, Suggested Specificati for Structures of Aluminium Alloys 6061-T6 and 6067-T6

5. The Aluminium Association, Specification for Aluminium Structures

SPENT FUEL STORAGE

Intent To Prepare Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel

From the early days of the nuclear power industry in this country, electric utilities planning to construct and operate light water nuclear power reactors contemplated that the used or spent fuel discharged from the reactors would be chemically reprocessed to recover the remaining quantities of fissile and fertile materials (uranium and plutonium), and that the materials so recovered would be recycled back into fresh reactor fuel. It was contemplated by the nuclear industry that spent fuel would be discharged periodically from operating reactors, stored in onsite fuel storage pools for a period of time to permit decay of radioactive materials contained within the fuel and to cool, and periodically shipped offsite for reprocessing. Typically, space was provided in onsite storage pools for about one and one-third nuclear reactor cores. Assuming a four-year reactor fuel reload cycle, such onsite storage pools were planned to hold an average of one year's discharge with sufficient remaining capacity to hold a complete core should unloading of all of the fuel from the reactor be necessary or desirable because of operational difficulties. Under normal operating conditions, an average of five years' discharge could be accommodated before the pools were filled.

Persons planning to conduct commercial reprocessing of spent reactor fuels provided sufficient storage capacity for the spent fuels at their facilities to allow some operational flexibility. Typically, space has been provided or planned for several spent fuel core reloads. Three commercial reprocessing plants have been planned for operation in the United States. The only such plant that has actually operated, Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for extensive alterations and expansion. There is a pending proceeding before the Nuclear Regulatory Commission (Commission) on NFS's application for a permit to construct these alterations and expansion (docket no. 50-201). The second plant, General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois, has never operated and is in a decommissioned condition. The third plant, Allied General Nuclear Services' (AGNS) proposed plant in Barnwell, South Carolina, is under construction and is the subject of pending proceedings before the Commission regarding the continuation, modification or suspension of the construction permit from an environmental protection standpoint, and the possible issuance of an operating license (docket no. 50-332), as well as a related matter (docket no. 70-1729).

On May 8, 1975, the Nuclear Regulatory Commission published a notice in the FEDERAL REGISTER setting forth its provisional views that, subject to consideration of comments, (1) a cost-benefit analysis of alternative safeguards programs should be prepared and set forth in draft and final environmental impact statements before a Commission decision is reached on wide-scale use of mixed oxide (recycle plutonium) fuels in light water nuclear power reactors, (2) there should be no additional licenses granted for use of mixed oxide fuel in light water nuclear power reactors except for experimental purposes, (3) with respect to light water nuclear power reactor fuel cycle activities which depend for their justification on wide-scale use of mixed oxide fuel in light water nuclear power reactors, there should be no additional licenses granted which would foreclose future safeguards options or result in unnecessary "grandfathering" and (4) the granting of licenses would not be precluded for fuel cycle activities for experimental and/or technical feasibility purposes.

In light of the status of the three planned commercial reprocessing plants in the United States, as outlined above, the earliest that spent fuel reprocessing could begin on a commercial basis, if authorized, would be late 1978. This assumes that the pending licensing proceedings are completed and licenses issued by this date. However, the spent fuel pools at a number of reactors may soon be filled, and still other reactors will have their pools filled before the end of 1978. Accordingly, even if limited reprocessing should begin in late 1976, there would still be a shortage in spent fuel storage capacity.

The existing pools at the GE and NFS reprocessing plants have some remaining marginal licensed storage capacity which may be able to accommodate the fuel discharges from some reactors; any increases planned at these plants may not be sufficient for industry in the future. Consequently, there is the possibility of a future shortage in licensed spent fuel capacity regardless of the outcome of the proceedings on the May 8th notice.

The Commission has not promulgated any regulation which specifies a given size for on-site reactor spent fuel pools; however, proposals by reactor licensees to significantly change the manner of spent fuel storage or spent fuel pool size would be subject to licensing review by the Commission. In the event that a particular on-site spent fuel pool should become filled, and no alternative form of spent fuel storage could be foun

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

April 15, 1983

Dockets Nos. 50-313
and 50-368

Mr. John M. Griffin, Vice President
Nuclear Operations
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

The Commission has issued the enclosed Amendments Nos. 76 and 43 to Facility Operating Licenses Nos. DPR-51 and NPF-6 for Arkansas Nuclear One, Units Nos. 1 and 2 (ANO-1 & 2). These amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 5, 1982, supplemented by letters dated February 17, 1983, March 3, 7, 10, 21, 22, 24, 28 and 29, 1983, and April 5 and 7, 1983.

These amendments allow an increase in the storage capacity for the ANO-1 spent fuel pool from 589 to 968 storage locations and of the ANO-2 spent fuel pool from 485 to 988 storage locations.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and Notice of Issuance/Negative Declaration are also enclosed.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 76 to DPR-51
2. Amendment No. 43 to NPF-6
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice/Negative Declaration

~~8304210192~~

Arkansas Power & Light Company

50-313, Arkansas Nuclear One, Unit 1
50-368, Arkansas Nuclear One, Unit 2

cc:

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO.1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-51

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated November 5, 1982, as supplemented February 17, 1983, and April 7, 1983, 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-51 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment Number and contain vertical lines indicating the areas of change.

Page

v
59
59a
59b
59c (new page)
59d (new page)
116
127

Amendment No. ~~52~~, 76

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable within 7 days prior to refueling operations. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.12 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.14 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.15 The spent fuel shipping cask shall not be carried by the Auxiliary Building crane pending the evaluation of the spent fuel cask drop accident and the crane design by AP&L and NRC review and approval. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.16 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.17 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.18 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

BASES

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.[1]

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

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.[2] Although the refueling boron concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.[3]; and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates, until the review specified in 3.8.15 is completed.

Specification 3.8.15 assures that the spent fuel cask drop accident cannot occur prior to completion of the NRC staff's review of this potential accident and the completion of any modifications that may be necessary to preclude the accident or mitigate the consequences. Upon satisfactory completion of the NRC's review, Specification 3.8.15 shall be deleted.

Specifications 3.8.16 and 3.8.17 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

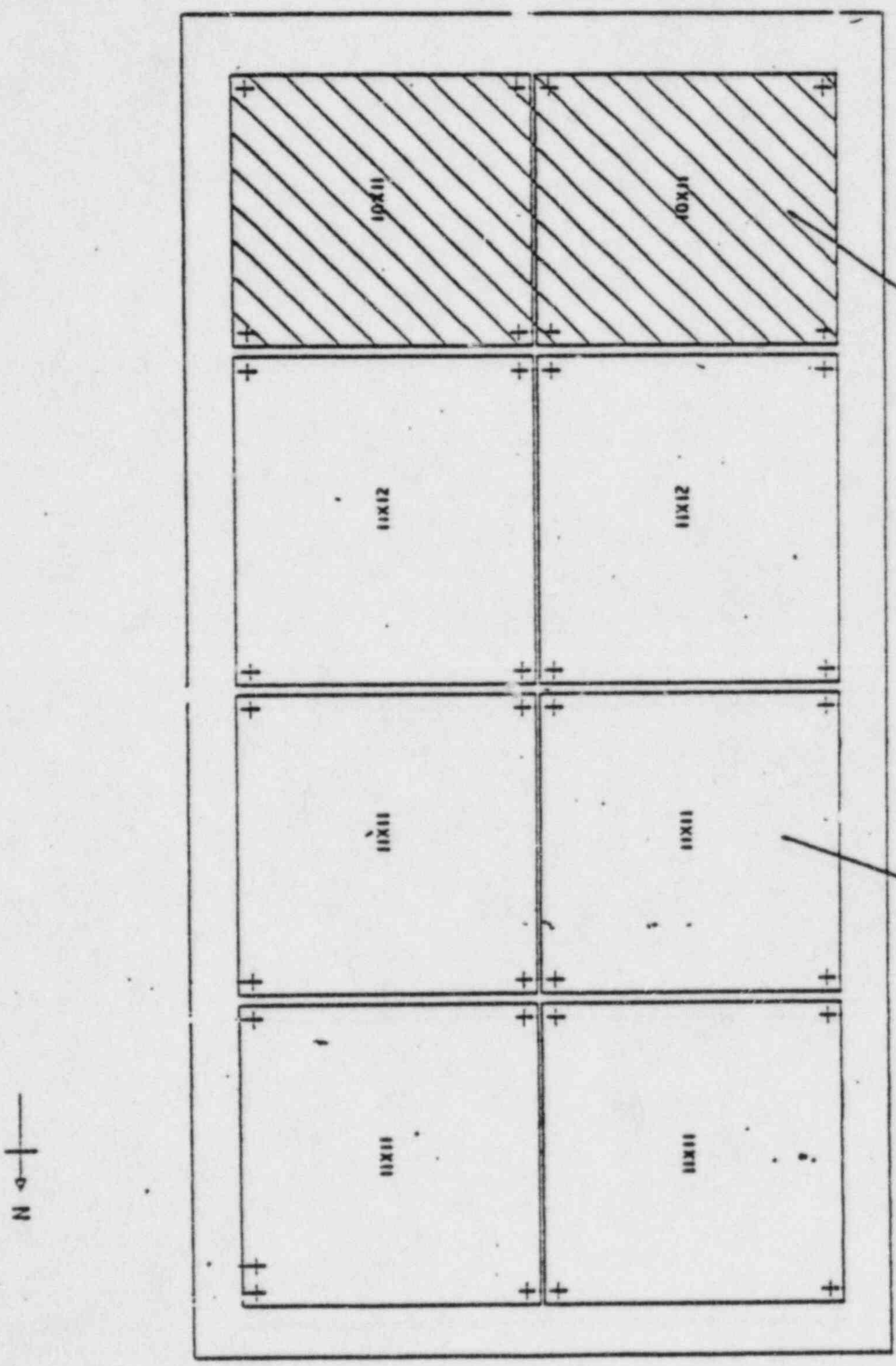
Specification 3.8.18 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3



REGION-1
220 LOCATIONS (10 X 22)

REGION-2
748 LOCATIONS (22 X 34)

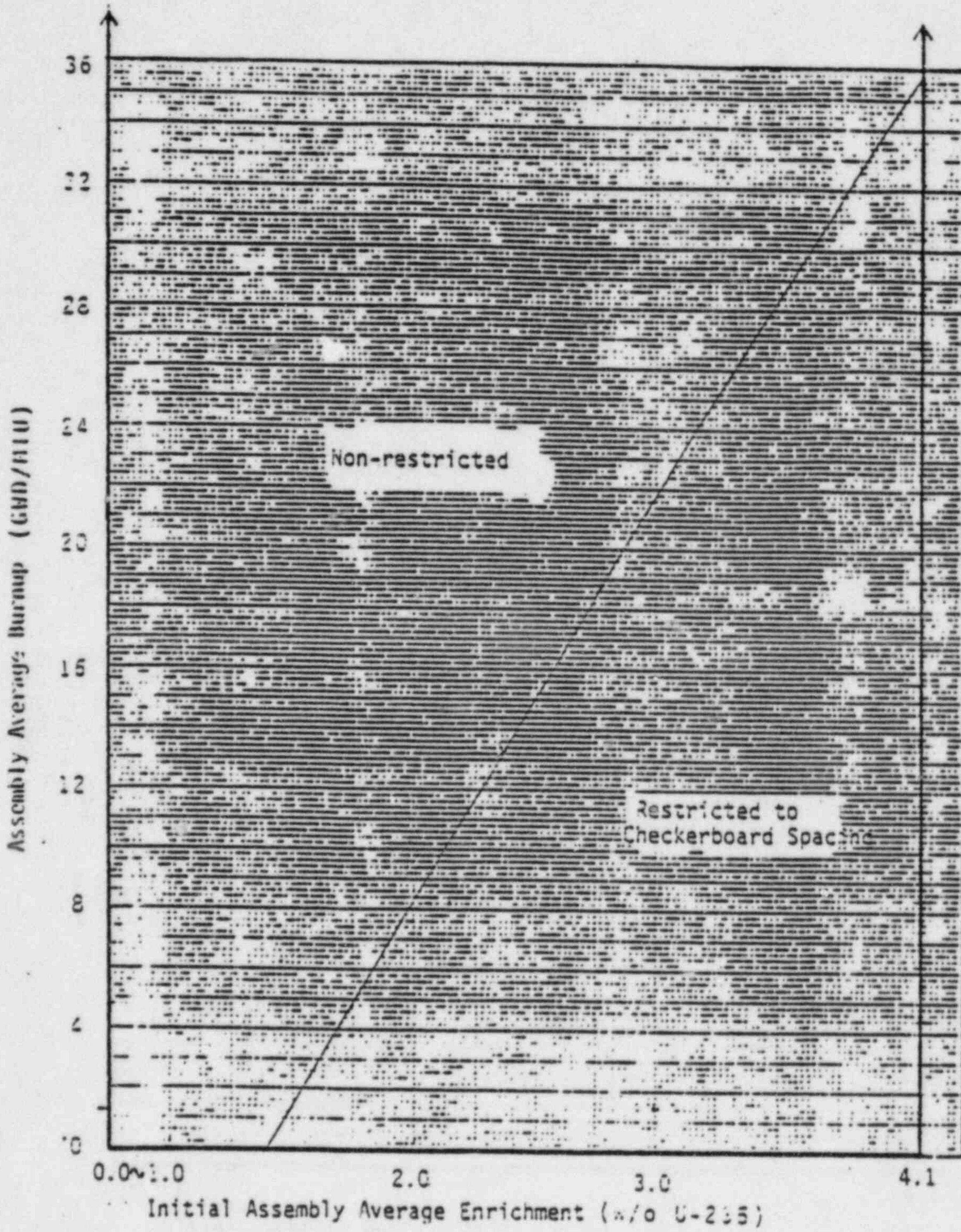
SPENT FUEL POOL ARRANGEMENT UNIT #1

Figure 3.8.1

FIGURE 3.8.2

MINIMUM BURNUP VS. INITIAL ENRICHMENT

FOR REGION 2 ENRICHMENT



5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 New Fuel Storage

1. Fuel assemblies are stored in racks of parallel rows, having a nominal center to center distance of 21 inches in both directions. This spacing is sufficient to maintain a K_{eff} of less than .9 even if flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent U235.

2. New fuel may be stored in the spent fuel pool or in its shipping containers.

5.4.2 Spent Fuel Storage

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water.

2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.

REFERENCES

FSAR, Section 9.6

a. The facility shall be placed in at least hot shutdown within one hour.

b. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.12.3.1.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. New and spent fuel storage.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the General Manager within 14 days of implementation.

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO.2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. NPF-6

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated November 5, 1982, as supplemented February 17, 1983, and April 7, 1983, 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-6 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are provided to maintain document completeness.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DECAY TIME AND SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.3.a The reactor shall be subcritical for at least 72 hours.

3.9.3.b In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for less than 175 hours, suspend all operations involving movement of more than 70 fuel assemblies from the reactor pressure vessel to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3.a The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.b The reactor shall be determined to have been subcritical for at least 175 hours by verification of the date and time of subcriticality prior to movement of the 71st irradiated fuel assembly from the reactor pressure vessel to the spent fuel pool.

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Exhausting through OPERABLE containment purge and exhaust system HEPA filters and charcoal adsorbers.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment.

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE at the following frequencies:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying with 31 days after removal that laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 39,700 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

b. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

c. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 39,700 cfm $\pm$ 10%.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm $\pm$ 10%.

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm $\pm$ 10%.

FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12.a Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.9.12.b Storage in Region 2 (as shown on Figure 3.9.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.9.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (Non-Restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable.

3.9.12.c The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

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

ACTION:

Suspend all actions involving the 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.

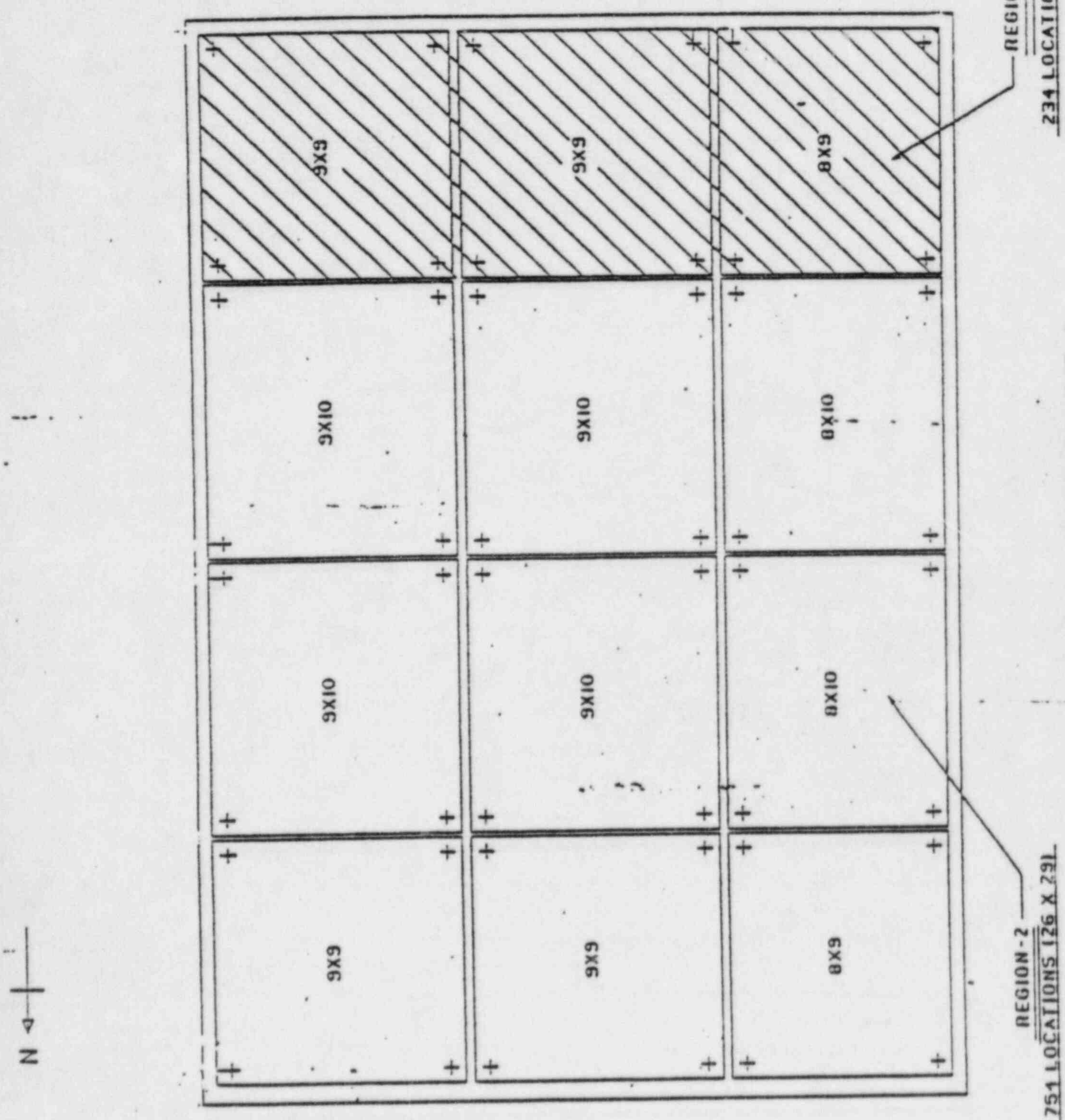
Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 1601 ppm, until such time as the boron concentration is increased to 1601 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.12.a Verify all fuel assemblies to be placed in the spent fuel pool had an initial enrichment of less than or equal to 4.1 w/o U-235 by checking the assemblies design documentation.

4.9.12.b Verify all fuel assemblies to be placed in Region 2 of the spent fuel pool are within the enrichment and burnup limits of Figure 3.9.2 by checking the assemblies design and burnup documentation.

4.9.12.c Verify at least once per 31 days the spent fuel pool boron concentration is greater than 1600 ppm.

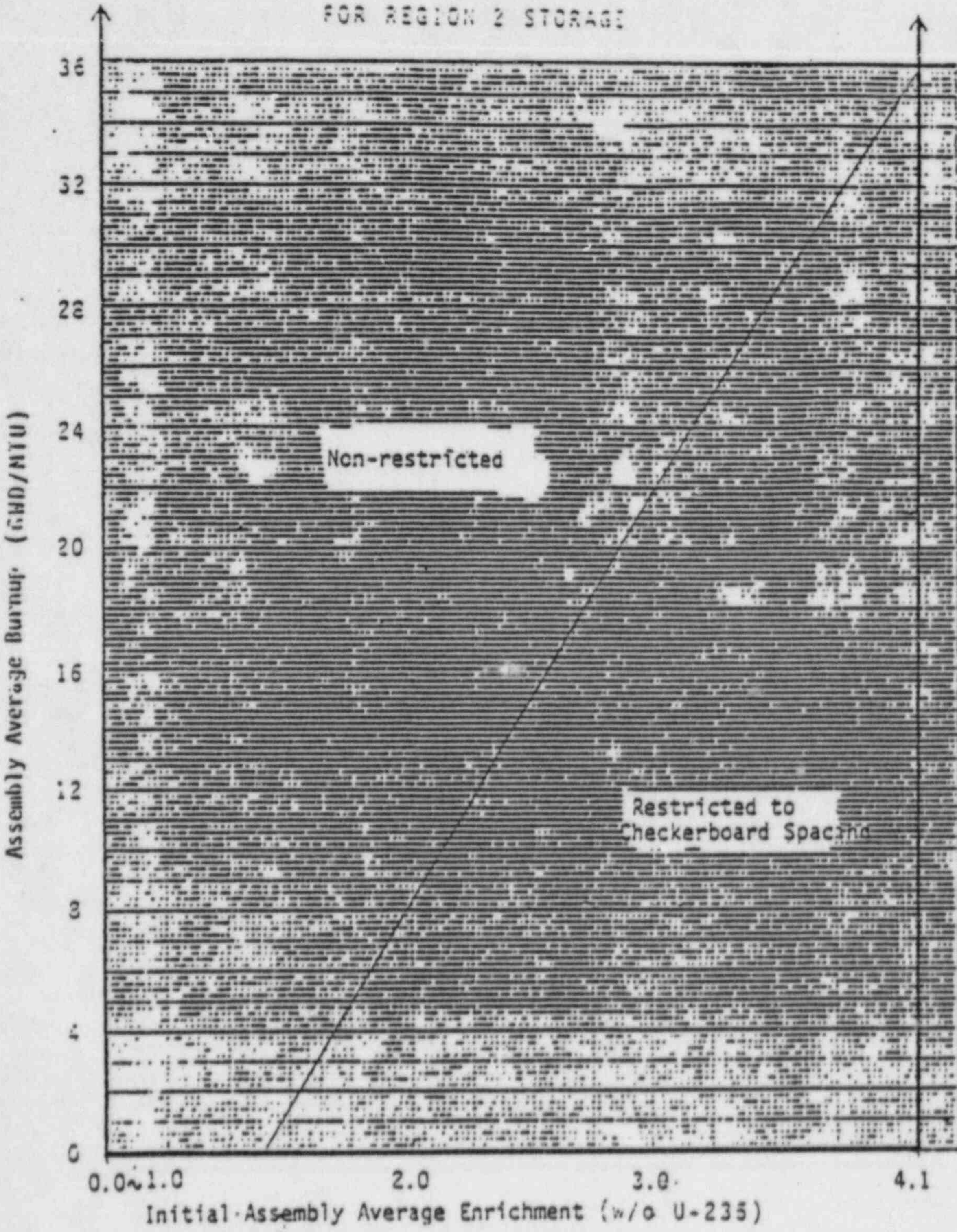


SPENT FUEL POOL ARRANGEMENT UNIT #2

FIGURE 3.9.2

MINIMUM BURNUP VS. INITIAL ENRICHMENT

FOR REGION 2 STORAGE



6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Vice-President, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants
NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Plant Safety Committee.

h. New and spent fuel storage.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the General Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

The minimum requirement for reactor subcriticality prior to movement of more than 70 irradiated fuel assemblies to the spent fuel pool ensures that sufficient time has elapsed to allow radioactive decay of the short lived fission products such that the heat generated will not exceed the cooling capacity of the spent fuel pool cooling system. This decay time and total assembly limitation is conservatively within the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the containment purge and exhaust system HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE OPERABILITY

The OPERABILITY requirements for the refueling machine ensure that: 1) the refueling machine will be used for movement of CEAs with fuel assemblies and that it has sufficient load capacity to lift a fuel assembly, and 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

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

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

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

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

3/4.9.11 FUEL HANDLING AREA VENTILATION SYSTEM

The limitations on the fuel handling area ventilation system ensure that all radioactive materials released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The operation of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.12 FUEL STORAGE

Region 1 of the spent fuel storage racks is designed to assure fuel assemblies of less than or equal to 4.1 w/o U-235 enrichment will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

Region 2 of the spent fuel storage racks is designed to assure fuel assemblies within the burnup and initial enrichment limits of Figure 3.9.2 will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

The requirement for 1600 ppm boron concentration is to assure the fuel assemblies will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in the event of a postulated accident.

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. DPR-51

AND

AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 & 2

DOCKETS NOS. 50-313 and 50-368

TABLE OF CONTENTS

1.0 Introduction

By letter dated November 5, 1982 (Ref 1), supplemented by References 2 through 14, Arkansas Power and Light Company (the licensee or AP&L) proposed amendments to Facility Operating Licenses Nos. DPR-51 and NPF-6 for Arkansas Nuclear One, Units Nos. 1 and 2 (ANO-1&2). The proposed amendments would revise the provisions in the Technical Specifications (TSs) to allow modifications in the spent fuel design for ANO-1&2 which would increase the spent fuel storage capabilities for ANO-1 from 589 spaces to 968 spaces and for ANO-2 from 485 spaces to 988 spaces. This expansion would be accomplished by replacing the existing spent fuel storage racks with new high density storage racks.

The proposed change would allow refueling capability through the 15th refueling scheduled for the spring of 1998 for ANO-1 and through the 14th refueling scheduled for the spring of 2000 for ANO-2. Present storage capacities would force the shutdown of ANO-1&2 in 1989 due to the inability to refuel.

As addressed below, we have evaluated the safety considerations associated with the proposed changes to the ANO-1&2 spent fuel storage designs. A separate Environmental Impact Appraisal addressing these changes has been prepared.

2.0 Evaluation

2.1 Criticality Considerations

For both ANO-1&2, the spent fuel storage racks are divided into two regions. Region 1 of each unit is designed to accommodate non-irradiated fresh fuel and is sized to permit core offloads. Storage in Region 2 for each unit is restricted by burnup and enrichment limits. Placement of fuel in Region 2 is determined by burnup calculations and controlled administratively by AP&L. Fuel which does not meet the burnup criterion may be placed in Region 2 in a checkerboard arrangement. In these cases, the vacant spaces adjacent to the assembly being inserted will be physically blocked to prevent inadvertent assembly insertion. In addition, the area designated will be subdivided from the normal storage in Region 2 by a row of vacant storage spaces. The criticality aspects of the design of each region are discussed separately below.

2.1.1 Region 1 Design

The Region 1 racks consist of individual stainless steel storage cells with a neutron absorbing material, Boraflex, attached to each cell. There are 234 fuel assembly storage locations with a 10.65 inch center-to-center spacing between assemblies for ANO-1 and 220 fuel assembly storage locations with a 9.8 inch center-to-center spacing between assemblies for ANO-2. The criticality analysis of the racks is performed with the state-of-the-art AMPX system of computer codes for neutron cross section generation and KENO IV for reactivity determination. KENO IV is a three-dimensional Monte Carlo theory computer code designed for reactivity calculations. These codes have been benchmarked against a set of 27 critical experiments in the range of pellet diameters, water-to-fuel ratios and U-235 enrichments that encompass the ANO-1 & 2 designs. This benchmarking led to the conclusion that the calculational model is capable of determining the multiplication factor of the Region 1 racks to within 1.3 percent in reactivity with a 95 percent probability at the 95 percent confidence level.

In the nominal case criticality calculation for Region 1, several worst case assumptions were made to account for some mechanical tolerance uncertainties. These included the most reactive eccentric assembly position within the can and reduced poison plate width. The effects of various other uncertainties and biases such as variation in water gap thickness and boron particle self-shielding are conservatively accounted for. Combining these uncertainties at the 95/95 probability/confidence level with the above-mentioned calculational uncertainty yields values of 0.9418 and 0.9448 for the multiplication factors of the Region 1 racks for ANO-1 & 2, respectively, when loaded with fuel assemblies of 4.1 weight percent U-235 enrichment at the pool temperature yielding the maximum reactivity and with the water (unborated) density conservatively taken as 1 gm/cc. This meets our acceptance criterion of less than or equal to 0.95 for this quantity.

We, therefore, conclude that any number of fuel assemblies of the Babcock & Wilcox (B&W) 15x15 design having enrichments no greater than 4.1 weight percent U-235 may be stored in Region 1 of the ANO-1 racks and that any number of fuel assemblies of the Combustion Engineering (CE) 16x16 design having enrichments no greater than 4.1 weight percent U-235 may be stored in Region 1 of the ANO-2 racks.

2.1.2 Region 2 Design

The Region 2 racks consist of a honeycomb structure of stainless steel cells surrounded by spacer pockets which are designed to accept poison inserts if future need arises. There are 748 fuel assembly storage locations with a 10.65 inch center-to-center spacing between assemblies for ANO-1, and 754 fuel assembly storage locations with a 9.8 inch center-to-center spacing between assemblies for ANO-2.

The same methods were used for the basic reactivity determination as were used in the Region 1 analysis. In addition, the LEOPARD/CINDER codes were used to calculate the isotopic compositions and neutron cross sections of the fuel as a function of burnup history and subsequent decay time. The TURTLE code is used to determine the reactivity equivalence of assemblies with different initial enrichments and burnups. Direct verification of the codes was not possible because no critical experiments have been done with assemblies having large burnups. Therefore, verification of various aspects of the calculation was undertaken. For example, the ability to calculate the isotopic composition of irradiated fuel was verified by comparing the LEOPARD/CINDER calculation to the measured results of irradiations performed on mixed oxide fuel in Saxton. Similar evidence was used to assess the fission product buildup uncertainty and its reactivity effect as well as the reactivity effect of the transuranium isotopes. The result of these uncertainties in addition to uncertainties due to the method, the nominal eigenvalue, construction and material tolerances, and asymmetric assembly positioning give a total 95/95 uncertainty of 2.48 percent reactivity change.

In order to establish burnup criteria for storage in Region 2 for each unit, a constant storage rack infinite multiplication factor (with minimum post-shutdown fission product inventory) contour is constructed as a function of burnup and initial enrichment using LEOPARD and TURTLE. This contour is based on a high enrichment endpoint of 4.10 weight percent and 36,000 MWD/MTU as shown in Figure 3.8.2 from the proposed ANO-1 TSs and in Figure 3.9.2 from the proposed ANO-2 TSs.

The final multiplication factors for Region 2 are determined using the same KENO IV method used for Region 1 with the conditions determined by the zero burnup intercept point in Figure 3.8.2 for ANO-1 and Figure 3.9.2 for ANO-2. In these cases, the intercept points are at 1.4 weight percent U-235. Therefore, the design mode for Region 2 for ANO-1 & 2 is an unirradiated assembly of 1.4 weight percent initial enrichment. LEOPARD and TURTLE are thus used only to calculate relative reactivities as a function of burnup while the KENO IV Monte Carlo method is used to determine the actual storage rack reactivity. The nominal case multiplication factors are calculated to be 0.8892 for ANO-1 and 0.9068 for ANO-2. Increasing these by the above calculated 95/95 uncertainty of 2.48 percent gives final Region 2 multiplication factors of 0.914 for ANO-1 and 0.9316 for ANO-2 which meet our acceptance criterion of less than or equal to 0.95. Based on our review, we conclude that any number of B&W design 15X15 fuel assemblies with burnups in the non-restricted region of Figure 3.8.2 may be stored in Region 2 of the ANO-1 spent fuel storage racks and that any number of CE design 16X16 fuel assemblies with burnups in the non-restricted region of Figure 3.9.2 may be stored in Region 2 of the ANO-2 spent fuel storage racks.

The multiplication factor for Region 2 is also determined assuming a checkerboard storage configuration with unirradiated fuel assemblies at 4.1 weight percent enrichment. The nominal multiplication factors determined by KENO IV are 0.9068 for ANO-1 and 0.8860 for ANO-2. Adding the 95/95 uncertainties due to the nominal eigenvalue, the method bias, tolerances in thickness and asymmetric assembly position results in values of 0.9402 for ANO-1 and

0.9169 for ANO-2 which meet our acceptance criterion of less than or equal to 0.95. Therefore, B&W design 15X15 fuel assemblies and CE design 16X16 fuel assemblies of any burnup and up to 4.1 weight percent enrichment may be stored in Region 2 of ANO-1 & 2 respectively in a checkerboard configuration with adjacent vacant spaces between stored assemblies.

2.T.3 Postulated Accidents

The effect of credible accidents has been considered and the most consequential one is the dropping of a single fuel assembly outside the rack between the periphery of the storage racks and the side walls of the pool. The effective multiplication factor remains below 0.95 for this accident with all uncertainties and biases included. The pool water was assumed to contain soluble boron for this analysis. This is permitted by the double contingency principle of ANSI N16.1-1975 "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," which states that two unlikely, independent, concurrent events are required to produce a criticality accident. We have accepted this principle in previous Safety Evaluations.

2.1.4 Administrative Procedures and Proposed TSs

ANO Administrative Procedure 1022.12, "Control and Accountability of Special Nuclear Materials," provides the controls to be used in determining the storage location for new and irradiated fuel in the spent fuel pools. The "Spent Fuel Pool Inventory Maps" reflect the additional storage locations in the reracking of the spent fuel pools and also the separation of the pools into two distinct regions. The procedure will include a description of the two regions and the method used to determine whether irradiated fuel should be placed in Region 1 or Region 2.

Figures 3.8.2 and 3.9.2 in the ANO-1 & 2 proposed TSs respectively describe the classification of each assembly as "Restricted" or "Non-Restricted" by comparing its burnup with its initial enrichment. The procedure will also include an evaluation for the guidelines pertaining to "Restricted" fuel when stored in Region 2 (i.e., by using a checkerboard pattern and separating these "Restricted" assemblies from the "Non-Restricted" assemblies by a vacant row of storage spaces). This procedure will require an independent check by two individuals classifying the irradiated fuel as "Restricted" or "Non-Restricted" and verifying the correct storage location considering the Region and the assembly identification number.

The proposed TSs governing the criticality aspects of the spent fuel pools for ANO-1 & 2 provide for limits on the initial enrichment of fuel assemblies, burnup limits, required boron concentration, limits on the calculated effective multiplication factors and physical blocks in the vacant spaces adjacent to any fuel assembly in Region 2 in the event a checkerboard storage configuration is deemed necessary.

2.1.5 Conclusions

We conclude that the proposed storage racks meet the requirements of General Design Criterion 62 with regard to criticality. This conclusion is based on the following considerations:

1. State-of-the-art calculation methods which have been verified by comparison with experiment have been used.

2. Conservative assumptions have been made about the enrichment of the fuel to be stored and the pool conditions.

3. Credible accidents have been considered.

4. Suitable uncertainties have been considered in arriving at the final value of the multiplication factor.

5. The final effective multiplication factor value meets our acceptance criterion.

We also conclude that the proposed modifications to the ANO-1 & 2 TSs are acceptable to allow operation with the proposed expansion of the spent fuel pools' storage capacities.

2.2 Spent Fuel Pool Cooling and Makeup

2.2.1 Introduction

Each ANO unit has an independent spent fuel pool and spent fuel pool cooling and cleanup system (SFPCS). The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain the water quality and clarity of the pool water. The ANO-1 SFPCS is composed of redundant trains, each train containing a pump and heat exchanger. The redundant trains can be cross-connected so that either pump can provide flow through either or both heat exchangers. The heat exchangers are cooled by the component cooling water system. The ANO-2 SFPCS is a closed loop system consisting of two half capacity pumps and one full capacity heat exchanger. The fuel pool water is drawn from the fuel pool near the surface and is circulated by the fuel pool pumps through the fuel pool heat exchanger where heat is rejected to the service water system.

The design of the storage pools is such that fuel will always be covered with water. Because of the locations of the fuel pool piping penetrations, the configuration of the pool and the use of siphon breaker vents, no incorrect operation or failure in the fuel pit cooling and refueling purification system could drain the fuel pool water level more than 4 feet below the normal level. The normal water level is 25 feet above the top of the fuel storage racks. In the event of a loss of the cooling system, makeup is available from the seismic Category I borated water storage tank or seismic Category I service water system. In addition, hose connections are available from the condensate tank and demineralized water supply.

The future refueling cycle for ANO-1 & 2 will be an 18-month period, and one-third of the core will be removed and stored in the spent fuel pool after each cycle. To limit the decay heat load, the one-third core will be removed from the reactor vessel and stored in the spent fuel pool 150 hours after reactor shutdown. In the event of a full-core discharge, the decay heat load will be limited by requiring a seven-day decay time after shutdown before core discharge. A full core contains 177 fuel assemblies.

2.2.2 Evaluation

To calculate the heat loads for the discharge of spent fuel to the pools, the licensee used Branch Technical Position ASB 9.2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." The maximum normal heat load that includes full core offload at the fifteenth refueling discharge was calculated to be 28.02 x 10 BTU/hr. for ANO-1 and 32.5 x 10 BTU/hr/ for ANO-2. The pool temperature at ANO-1 is maintained below 120 F by operating any combination of two pumps and two heat exchangers for the normal heat load and at or below 150 F for the maximum normal conditions (full core offload). Upon failure of one pump or heat exchanger for the normal condition, sufficient cooling capacity remains to maintain bulk pool temperature below 135 F. Similarly, at ANO-2, the pool temperature is maintained under 120 F by recirculating spent fuel cooling water from the spent fuel pool through the parallel arranged pumps and a heat exchanger and back to the pool. For maximum normal conditions (full core offload), the pool temperature is maintained at or below 150 F, which meets the guidelines of Standard Review Plan Section 9.1-3, "Spent Fuel Pool Cooling and Cleanup

System." The American National Standard 57.2, " Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," indicates that the maximum pool temperature should not exceed 150 F under normal operating conditions with all storage racks full. The design, therefore, also meets this standard.

To verify that natural circulation of the pool water for the proposed expanded rack configuration provides adequate cooling of all fuel assemblies in the event of a loss of external cooling, the licensee performed a thermal-hydraulic analysis. In the event of the complete failure of the spent fuel cooling system, for the maximum normal heat load, there is at least four hours available before boiling occurs. The maximum boiloff rate is 50 to 60 gpm. Each of the two assured seismic Category I borated makeup water sources can be initiated in the required time. Sufficient makeup rates are also available from the seismic Category I service water system, condensate tank or demineralized water supply.

2.2.3 Conclusion

We have reviewed the calculated decay heat values and conclude that the heat loads are consistent with the Branch Technical Position ASB 9.2 and therefore, are acceptable. The SFPCS performance has been reviewed, and we conclude that the pool cooling is adequate. The available makeup systems, their respective makeup rates and the time required before makeup if needed have been reviewed and found acceptable. Based on the above, we conclude that the SFPCSs are acceptable for the proposed expansions.

2.3 Installation of Racks and Load Handling

2.3.1 Description

The proposed spent fuel storage modifications will provide storage locations for 968 fuel assemblies for ANO-1 and 988 fuel assemblies for ANO-2. The spent fuel storage racks are divided into two regions. Region I is designed to accommodate non-irradiated fresh fuel and is sized to permit core offloads. Storage in Region 2 is restricted by burnup and enrichment limits. There is no physical barrier between the two regions. Each fuel assembly will be stored in a double walled storage cell of type 304 stainless steel. The annular spaces between the double walls of the cells contain B C (Boroflex) neutron absorber elements positioned at the rack height corresponding to the active fuel length of the fuel assemblies. The individual storage cells are welded into rack arrays. At ANO-1, the storage racks will have three basic module configurations with dimensions of 10 x 11, 11 x 12 and 11 x 11 feet and weigh 27,500 lbs., 19,500 lbs. and 18,000 lbs. respectively. There will be two 10 x 11 modules, two 11 x 12 modules and four 11 x 11 modules. Similarly, at ANO-2 the storage racks will have a module configuration with dimensions of four 9 x 9, two 8 x 9, four 9 x 10 and two 8 x 10 feet. These modules will weigh from 13,000 lbs. to 20,300 lbs. The above configuration maintains cell pitch of 10.65 inches at ANO-1 and 9.8 inches at ANO-2 and prevents placement of a fuel assembly anywhere other than a design location.

The proposed neutron absorber fuel racks are designed to seismic Category I criteria. Structural and seismic analyses have been performed by the licensee to verify that the rack design is adequate to withstand normal operating, seismic and accident loading conditions.

2.3.2 Rack Handling and Installation

The review of heavy load handling at ANO-1 & 2 is being conducted as part of the ongoing generic review initiated by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The results of that review will be reported as part of Multiplant Action Item C-10. The evaluation provided herein is limited to the heavy load handling activities associated with the proposed spent fuel storage modifications.

Each unit has one seismic Category I overhead crane in the auxiliary building which will be used for removing the existing rack modules and lowering the new modules into the pool. The licensee has stated in its November 5, 1982 submittal that "no loads exceeding 2000 lbs. will be allowed over the fuel assemblies at any time." The TSs for ANO-1 & 2 also prohibit the travel over fuel assemblies in the storage pool of loads in excess of 2000 lbs. Since the weight of a rack module is much greater than 2000 lbs., we conclude that the rack modules will not be carried over the fuel assemblies and that there is reasonable assurance that an accident impacting assemblies in the pool will not occur. All movement of spent fuel racks will be controlled by written administrative procedures which will prohibit movement of the racks over locations in the pool where fuel is stored.

The licensee indicated that the movement of all loads into and out of the auxiliary building associated with this modification will be accomplished with the single-failure proof cask crane and double rigging to assure that a single failure will not result in an unanalyzed load-drop event. The licensee has committed to establish a program for installation and use of slings which complies with the criteria contained in ANSI B30.9-1971. In NUREG-0612, we concluded that this is acceptable.

The licensee also stated in response to NUREG-0612 that all crane operators and signalmen will be trained in accordance with ANSI B30.2-1976, and no exceptions from the standards are taken regarding training, qualification or operator conduct.

2.3.3 Conclusion

We have reviewed the described load handling operations and equipment needed for the spent fuel rack modifications. We conclude that the lifting devices and other apparatus used for the handling of the storage racks are acceptable.

2.4 Structural Design

2.4.1 Introduction

Both units at ANO are pressurized water reactors (PWRs). The spent fuel pools are similar right and lefthand arrangements. The pools are elevated with the top of the pools at the fueling floor level, elevation 404 feet. The inside bottom of the pools is at elevation 362 feet. The top of the slab-on-grade is at elevation 335 feet. The approximate inside dimensions of the pools are:

	ANO-1	ANO-2
Depth	42 ft.	42 ft.
Length	44 ft.	32.75 ft.
Width	23 ft.	23 ft.

The pool structures are reinforced concrete with floor thickness of about 5.15 feet and walls of various thickness from 4 to 6 feet. The outside walls of the pools are generally continuous to the foundation mat. These walls support the bottom slab of the pool.

Each pool is lined with a continuous, welded, watertight, 3/16 inch thick stainless steel plate.

The new racks are stainless steel "egg-crate" structures. The 9 cell by 9 cell rack is approximately 16 feet high by 7.4 feet long by 7.4 feet wide. The cells of the egg-crate are fabricated of cold-formed gage thickness material. These cells are supported by a heavy welded base and by a welded structural grid near the top of each rack. The racks are each free standing on the pool floor on four corner leveling pads.

2.4.2 Applicable Codes, Standards and Specifications

Structural material of the racks conform to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. Computed stresses were compared with the ASME Code, Section III, Subsection NF. Load combinations and acceptance criteria for racks were compared with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979 (hereafter referred to as the "NRC Position").

The pool structures were evaluated in accordance with the requirements of ACI 349-80 for load combinations based on the NRC Standard Review Plan, NUREG-0800, Section 3.8.4.

2.4.3 Loads and Load Combinations

Load and load combinations for the racks and the pool structures were reviewed and found to be in agreement with the applicable portions of the NRC Position.

2.4.4 Seismic and Impact Loads

Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plants at the licensing stage. This was based on a 0.2g safe shutdown earthquake (SSE) and a 0.1g operating basis earthquake (OBE). Acceleration in the verical direction was computed as being two-thirds of horizontal acceleration. Damping values for the seismic analysis of the racks and the pool structures were taken as two percent for OBE. Rack/fuel bundle interactions were considered in the structural analysis. The SSE load was computed as twice the OBE load.

Loads due to a fuel bundle drop accident were considered in a separate analysis for such an occurrence.

The postulated loads from such events were found to be acceptable.

2.4.5 Design and Analysis of the Racks

A non-linear, time history analysis was performed on a two dimensional model of the rack. This model included consideration of sliding and tipping of the racks as well as potential rack-to-fuel bundle impacts. The model consisted of spring, mass, damping and gap elements arranged to simulate the rack and fuel. Hydrodynamic effects were considered. Estimates of sliding and tipping of the racks were taken from the analysis and used in combination with thermal considerations to establish minimum limiting gaps between the racks in order to preclude rack-to-rack impacts.

A linear, response spectrum, finite element analysis was performed to design/verify the rack structure.

The rack structural design produced calculated stresses for the rack components which were within allowable limits. The racks were found to have adequate margins against sliding and tipping.

An analysis was conducted to assess the potential effects of a stuck fuel assembly causing an uplift load on the racks and a corresponding downward load on the lifting device as well as a tension in the fuel bundles. Resulting stresses were found to be within acceptance limits.

2.4.6 Seismic Analysis of the Pool Structure

A structural analysis of the reinforced concrete pool structures was conducted by the licensee, and it was found that each pool structure is adequate to withstand the effects of added loads due to the new racks under seismic loads. The analysis consisted of a detailed finite element examination of the pools including thermal and seismic loads as well as other applicable loads. No overstress conditions exist in the pool structures or liners for the proposed installations.

2.4.7 Conclusion

It is concluded that the proposed rack installations will satisfy the requirements of 10 CFR 50, Appendix A, GDC 2, 4, 61 and 62 as applicable to structures, and are therefore acceptable.

2.5 Materials

2.5.1 Materials Description

The proposed spent fuel storage racks have been fabricated of type 304 stainless steel, which is used for all structural components. The storage pool in each of the two units is divided into two regions. Region 1 in each case utilizes Boraflex as a neutron absorber material, attached to the active portion of each fuel assembly cell by a thin wrapper which is welded in place. Placement of the wrapper provides for venting the Boraflex to the pool environment, thereby eliminating potential pressure buildup, for example by radiolysis of entrained water vapor. Depending on criticality requirements, Boraflex is deployed on either all four sides, three sides or two sides of a cell. Region 2 features storage racks consisting of cells assembled in a checkerboard pattern, producing a honeycomb-type structure. Each cell has attached to its outer wall a stainless steel wrapper plate creating a pocket opened at the top and bottom. The spacer pockets are designed to accept poison inserts if future need arises. The type 304 stainless steel rack modules have been welded and inspected by nondestructive examinations performed in accordance with the applicable provisions of ASME Boiler and Pressure Vessel Code, Section III (and therefore, by reference, Section IX).

2.5.2 Chemical Compatibility

The spent fuel pools of ANO-1 and 2 are fabricated of materials that will have good compatibility with the borated water chemistry of the spent fuel pool. The corrosion rate of type 304 stainless steel in this water is sufficiently low to defy our ability to measure it. Since all materials in the pools are stainless steel, no galvanic corrosion effects are anticipated. No instances of corrosion of stainless steel in spent fuel pools containing boric acid have been observed throughout the country (Ref. 15). Boraflex has been shown to be resistent to radiation doses in excess of any anticipated in the spent fuel pools of ANO-1 & 2 (Ref. 16). The venting of the cavities containing the Boraflex to the spent fuel pool environment will ensure that no gaseous buildup will occur in these cavities that might lead to distortion of the racks. The Codes and Standards used in fabricating and inspecting these new fuel storage racks should ensure their integrity and minimize the likelihood that any stress corrosion cracking will occur during service. AP&L has described a materials surveillance program which would reveal instances of deterioration of the Boraflex during the life of the new spent fuel racks. The monitoring program consists of a series of eight jacketed poison coupons which duplicate the condition of Boraflex encased in the poison canisters. These coupons are to be hung alongside the high density racks and will be subjected to the neutron, gamma and heat fluxes. Sufficient coupons are included to permit examination of a sample on inspection intervals of 1 to 5 years over the life of the facilities. An additional strap of eight coupons will be suspended adjacent to the most recently discharged fuel element at each off-loading and examined at each subsequent off-loading. By an evaluation of these specimens, an accelerated testing of environmental effects will be obtained, simulating within an eight-year period the effects upon the normally exposed poison material during a 40-year period.

This monitoring program will ensure that, in the unlikely situation that the Boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

2.5.3 Conclusion

From our evaluation as discussed above, we conclude that the corrosion that will occur in the spent fuel pools will be of little significance during the remaining life of the units. Components of the spent fuel storage pools are constructed of alloys which are known to have a low differential galvanic potential between them, and that have performed well in spent fuel storage pools at other PWR sites where the water chemistry is maintained to comparable standards to those in force at ANO-1 & 2. The proposed materials surveillance program is adequate to provide warning in the unlikely event that deterioration of the neutron absorbing properties of the Boraflex will develop during the design life of the racks. Therefore, with the selection of the materials, we believe that no significant corrosion should occur in the spent fuel storage racks for a period well in excess of the design life of the units.

We therefore conclude that the compatibility of the materials and coolant used in the spent fuel storage pools is adequate based on tests, data, and actual service experience in operating reactors. We find that the selection of appropriate materials by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, by preventing criticality by maintaining structural integrity of components, and is therefore acceptable.

2.6 Spent Fuel Pool Cleanup System

2.6.1 Introduction

The spent fuel pool cleanup systems for ANO-1 and 2 consist of a demineralizer for each unit (mixed bed resin), filters, and associated piping, valves and fittings. The systems are designed to remove corrosion products, fission products, and impurities from the pool water. Pool water purity is monitored by monthly chemical and radiochemical analyses. Demineralizer resin will be replaced on the basis of an increase in differential pressure or when pool water samples show reduced decontamination effectiveness. However, these resins are routinely changed on an annual basis as a preventive measure even though they may not show reduced decontamination effectiveness. The licensee indicated that no change or equipment addition to the spent fuel pool cleanup systems is necessary to maintain pool water quality for the augmented storage facilities.

2.6.2 Evaluation

The spent fuel pool cleanup systems have been reviewed in accordance with Section 9.1.3 of the Standard Review Plan (NUREG-0800, July 1981).

Past experience showed that the greatest increase in radioactivity and impurities in spent fuel pool water occurs only during refueling and spent fuel handling. The refueling frequency, amount of the core to be replaced for each fuel cycle, and frequency of operating the spent fuel pool cleanup systems at ANO-1 and 2 are not expected to increase as a result of expansion of the spent fuel pools. There is no reason to believe that the chemical and radionuclide composition of the spent fuel pool waters will change as a result of the proposed modifications. Past experience also indicated that there is not any significant leakage of fission products from spent fuel stored in pools after the fuel has cooled for several months. Thus, the increased quantity of spent fuel to be stored at ANO-1 and 2 will not contribute significantly to the amount of radioactivity from fission products in the spent fuel pool waters.

On the basis above, we determined that the proposed expansion of the spent fuel pools at ANO-1 & 2 will not affect the capability and capacity of the spent fuel pool cleanup systems. Accordingly, no change to the present systems is required. More frequent replacements of the filters or demineralizer, required when the decontamination effectiveness is reduced, can offset any potential increase in radioactivity and impurities in the pool water as a result of the expansion. Thus, we have determined that the existing spent fuel pool cleanup systems with the proposed fuel storage expansion (1) provide the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool waters, and thus meet the requirements of General Design Criterion 61 in Appendix A to 10 CFR Part 50, as it relates to appropriate filtering systems for fuel storage; (2) are capable of reducing occupational exposures to radiation by removing radioactive products from the pool waters, and thus meet the requirements of Section 20.1(c) of 10 CFR Part 20, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confine radioactive materials in the pool waters into the demineralizer and filters, and thus meet Regulatory Position C.2.f(2) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the source; and (4) remove suspended impurities from the pool water by filters, and thus meet Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing cruds from fluids through physical action.

2.6.3 Conclusion

On the basis of the above evaluation, we conclude that the spent fuel pool cleanup systems meet GDC 61, Section 20.1(c) of 10 CFR Part 20 and the appropriate sections of Regulatory Guide 8.8 and, therefore, are acceptable for the proposed expansion of the spent fuel pools.

2.7 Occupational Radiation Exposure

We have reviewed the radiation protection aspect of the licensee's plans to modify the spent fuel pools for ANO-1 & 2.

The licensee has estimated 16 man-rems will be the collective occupational dose in replacing the ANO-1 & 2 spent fuel storage racks. This collective dose estimate includes detailed breakdown of exposure to individuals performing specific tasks for each phase of the following operations: decontamination, rack removal, clean-up and disposal and new rack installation. The licensee has also outlined measures that will be taken to ensure personnel exposure for divers working in the spent fuel pools is ALARA. Lessons learned from previous re-rack experience are also included in the program.

The licensee does not expect any significant increase in dose rates due to the buildup of crud along the sides of the pools. If crud buildup eventually becomes a major contributor to pool dose rates, measures will be taken to reduce such dose rates. The purification system for the pools includes filters and demineralizers to remove crud and will be operating during the modifications of the pools.

The licensee has presented four alternative plans for removal and disposal of the old racks. These are (1,2) burial with or without volume reduction; (3) decontaminate to releasable criteria of Regulatory Guide 1.86 and disposal; (4) to have an outside vendor chemically decontaminate and dispose of the intact racks. The disposal methodology will follow ALARA guidelines for each of the alternatives.

The licensee has an ALARA committee, which reviews all work in radiological controlled areas when the estimated collective dose for any job will exceed 1 man-rem. Some of the actions that will be taken by the licensee to assure that occupational doses during each task of the pool modifications will be ALARA are:

1. A health physicist and diving supervisor will be in direct communication with the divers at all times during the re-racking to monitor for excessive exposure by utilizing portable or hand-held radiation monitoring instruments. The dose rates will not be permitted to exceed 1 rem/hr whole body.

2. Personnel monitoring devices will be used by all personnel working in the radiologically-controlled area. Additional monitoring of the underwater divers will be done by multiple whole body TLDs and extremity TLDs.

3. Personnel shall be required to wear appropriate protective clothing as determined by the health physicist to preclude contamination.

4. As the racks are pulled out of the water, they will be washed.

5. Area radiation monitors will be used to alarm on a high radiation signal. Actual dose rates can be read locally and in the control room.

6. A portable filtered water vacuum system will be available to remove loosely deposited contamination from the fuel rack surfaces, pool floor and walls near divers' working areas to reduce the radiation exposure.

7. Contamination control measures will be used to prevent the spread of contamination and to protect personnel from internal exposure from radioactive material

8. Underwater radation surveys will be performed in all areas where divers must work or have the need for access to the work area. An underwater radiation monitoring instrument will be used to perform dose rate measurements in the pools.

Based on our review of the ANO-1 & 2 spent fuel pool modification description and relevant experience from other operating reactors that have performed similar modifications, we conclude that the licensee's modifications can be performed within the limits of 10 CFR Part 20 and in a manner that will maintain doses to workers ALARA.

We have estimated the increment in occupational dose during normal operations, after the pool modifications, resulting from the proposed increase in stored fuel assemblies. The spent fuel assemblies contribute a negligible amount to dose rates in the pool areas because of the depth of water shielding the fuel; the major source of dose rate is the radionuclide concentrations in the pool water. The mose significant contributor to the radionuclides is the movement of fuel rather than the number of fuel assemblies in the pools. Thus the additional assemblies will add a negligible amount to area dose rates. Based on present and projected operations in the spent fuel pool areas and experience from similar modifications, we estimate that the proposed modifications should add less than one percent to the total annual occupational radiation dose to plant personnel. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational doses within the limits of 10 CFR Part 20 and ALARA.

On the basis of the above, we have determined that the dose to personnel will be maintained within the limits of 10 CFR 20, "Standard for Protection Against Radiation", and as low as is reasonably achievable, and therefore, the licensee's occupational dose control program is acceptable.

2.8 Radioactive Waste Treatment

Each unit contains waste treatment systems designed to collect and process the gaseous, liquid, and solid wates that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation Reports (SERs) for Arkansas Nuclear One, Units Nos. 1 and 2, dated June 1973 and November 1977, respectively. The proposed modifications will not result in any significant additional radwastes that will need to be processed. Therefore, there will be no change in the waste treatment systems or in the conclusions given in Section 11.0 of the SERs because of the proposed modifications.

2.9 Radiological Consequences of Rack Module Assembly Drop, Cask Drop and Fuel Handling Accidents

2.9.1 Introduction

We have reviewed the licensee's plans for the expansion of the storage capacity of the spent fuel pools at ANO-1 & 2 regarding radiological consequences of rack module drop, cask drop and fuel handling accidents. The review was conducted according to the guidance of Standard Review Plans 15.7.4 and 17.7.5, and Regulatory Guide 1.25.

2.9.2 Evaluation and Findings

Rack Module Assembly Drop Accident

The overhead cranes in the auxiliary buildings at ANO-1 & 2 will be used for removing the existing rack modules and lowering the new modules into the pools. The licensee has stated in Section 8.1, Rack Modules Assembly Handling Considerations, of the November 5, 1982 submittal that "no loads exceeding 2000 lbs. will be allowed over the fuel assemblies at any time." The TSs for ANO-1 & 2 also prohibit the travel over fuel assemblies in the storage pool of loads in excess of 2000 lbs. Since the weight of a rack module is much greater than 2000 lbs., we conclude that the rack modules will not be carried over the fuel assemblies and that there is reasonable assurance that an accident impacting assemblies in the pools would not occur. The assessment of the radiological consequences of a rack module assembly drop accident is not required.

Fuel Handling Accident

The maximum weight of loads which may be transported over spent fuel in the pool is limited by TSs to that of a single assembly (2000 lbs.). The proposed spent fuel pool modifications do not increase the radiological consequences of fuel handling accidents considered in our SERs of June 1973 (ANO-1) and November 1977 (ANO-2), since this accident would still result in, at most, the release of the gap activity of one fuel assembly due to the limitation on the available impact kinetic energy.

Cask Drop Accident

In the evaluation of the cask drop accident, the licensee states in the November 5, 1982 submittal that the administrative procedures prevent a spent fuel cask from being moved over the spent fuel pools. We conclude that the proposed spent fuel pool modifications do not affect the result of the cask drop accident considered in the SERs.

2.9.3 Conclusion

Based upon the above evaluation, we conclude that the likelihood of a rack module assembly drop accident is sufficiently small - since the rack module assembly will not be allowed over the fuel at any time - that this accident need not be considered. Also, a fuel handling accident involving a dropped assembly or cask would not be expected to result in radionuclide releases leading to offsite radiological consequences exceeding those of the fuel handling accident evaluated in our SERs of June 1973(ANO-1) and November 1977 (ANO-2); that is, doses would be well within 10 CFR Part 100 values. We conclude therefore, that the proposed modifications are acceptable.

3.0 Conclusions

Based on our review, we conclude that the proposed modified fuel storage designs of 968 fuel assemblies for ANO-1 and 988 fuel assemblies for ANO-2 of 4.1 weight percent U-235 enrichment meet the requirements of General Design Criterions 2, 4, 61 and 62 of Appendix A to 10 CFR Part 50 and are, therefore, acceptable. Based on our review, we have determined that the proposed TS changes for ANO-1 & 2 are acceptable.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE

SPENT-FUEL STORAGE POOLS

FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 AND 2

DOCKET NOS. 50-313 AND 50-368

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1.0 INTRODUCTION

The storage capacity of the spent fuel pools at Arkansas Nuclear One, Unit 1 (ANO-1) and Unit 2 (ANO-2) is 589 fuel assemblies for ANO-1 and 485 fuel assemblies for ANO-2. These limited storage capacities were in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a few years and then shipped offsite for reprocessing and recycling of the fuel.

Commercial reprocessing of spent fuel has not developed as had been originally anticipated. In 1975 the Nuclear Regulatory Commission directed the staff to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement was to consider alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August 1979. In the FGEIS, consistent with long range policy, the storage of spent fuel is considered to be interim storage, to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of onsite fuel storage capacity by modification of the existing spent fuel pools. Since the issuance of the FGEIS, applications for approximately 95 spent fuel pool capacity expansions have been received and 81 have been approved. The remaining 14 are still under review. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommends that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing spent fuel pools, the FGEIS discusses in detail other spent fuel storage alternatives. The finding of the FGEIS is that the environmental impact costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of various alternatives reflect the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, that of shutting down the reactor when the existing spent fuel storage capacity is filled, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Impact Appraisal (EIA) addresses only the specific environmental concerns related to the proposed expansion of the Arkansas Nuclear One (ANO) spent fuel storage capacity. This EIA consists of three major parts, plus a summary and conclusion. The three parts are: (1) descriptive material, (2) an appraisal of the environmental impact of the proposed action, and (3) an appraisal of the environmental impact of postulated accidents. Additional discussion of the alternatives to increasing the storage capacity of existing spent fuel pools is contained in the FGEIS.

1.1 Description of the Proposed Action

By application dated November 5, 1982 and supplemented by Reference 2 through Reference 14, Arkansas Power & Light Company proposed amendments to the Arkansas Nuclear One Facility Operating License Nos. DPR-51 (Unit 1) and NPF-6 (Unit 2). The proposed amendments would allow increases in the storage capacity of the ANO-1 Spent Fuel Pool (SFP) from 589 fuel assemblies to a maximum of 968 fuel assemblies and the ANO-2 SFP from 485 fuel assemblies to a maximum of 988 fuel assemblies. The increases are to be accomplished by reracking the SFPs with high density storage racks.

The environmental impacts associated with the operations of ANO-1 and ANO-2 were considered in the NRC's Final Environmental Statements (FESs) issued in February 1973 for ANO-1 and June 1977 for ANO-2. The purpose of this EIA is to evaluate any additional environmental impacts which are attributable to the proposed increases in the SFP storage capacity at ANO.

1.2 Need for Increased Storage Capacity

ANO-1 is a Babcock & Wilcox pressurized water reactor (PWR) unit and its reactor core contains 177 fuel assemblies. ANO-2 is a Combustion Engineering PWR unit and its reactor core also contains 177 fuel assemblies. The present SFP storage capacity of ANO-1 and ANO-2 is 589 fuel assemblies and 485 fuel assemblies, respectively. The licensee's projected SFP capacity requirements are presented in Figures 11.1 and 11.2 of Reference 1. Based on these projections, ANO-1 and ANO-2 will lose full core discharge capacities in 1986 and normal reload discharge capacities will be lost in 1989 for both units. Therefore, additional SFP storage capacity is required if ANO is to operate beyond the year 1989. It should be noted that the facility operating licenses for ANO-1 and ANO-2 expire in the year 2008 and in the year 2012, respectively.

1.3 Fuel Reprocessing History

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for alterations and expansion; in September, 1976, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York are licensed to store spent fuel. The storage pool at West Valley is not full, but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractual arrangements with NFS. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris operation was renewed for another 20 years; however, GE is also not accepting any additional spent fuel for storage at this facility.

2.0 FACILITY

The principal features of the spent fuel storage and handling at ANO as they relate to the proposed modifications are described here to aid understanding of the evaluations provided in subsequent sections of this EIA.

2.1 Spent Fuel Pool (SFP)

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core; also, they have a high thermal output. The SFP is designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling. The ANO-1 SFP is approximately 23 ft. wide x 44 ft. long x 42 ft. deep and the ANO-2 SFP is approximately 23 ft. wide x 32 3/4 ft. long x 42 ft. deep. The SFP structures are reinforced concrete lined with a continuous, watertight stainless steel plate.

2.2 Spent Fuel Pool Cooling and Cleanup System

Each ANO Unit has an independent spent fuel pool and spent fuel pool cooling and cleanup system. The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain the water quality and clarity of the pool water. The ANO-1 spent fuel pool cooling system is composed of redundant trains, each train containing a pump and heat exchanger. The redundant trains can be cross-connected so that either pump can provide flow through either or both heat exchangers. The heat exchangers are cooled by the component cooling water system. The ANO-2 spent fuel pool cooling system is a closed loop system consisting of two half capacity pumps and one full capacity heat exchanger. The fuel pool water is drawn from the fuel pool near the surface and is circulated by the fuel pool pumps through the fuel pool heat exchanger where heat is rejected to the service water system.

Each spent fuel pool cleanup system consists of a demineralizer (mixed bed resin), filters, and associated piping, valves and fittings. The systems are designed to remove corrosion products, fission products, and impurities from the pool water. Pool water purity is monitored by monthly chemical and radiochemical analyses. Demineralizer resin will be replaced on the basis of an increase in differential pressure or when pool water samples show reduced decontamination effectiveness. However, these resins are routinely changed on an annual basis as a preventive measure even though they may not show reduced decontamination effectiveness. The licensee indicated that no change or equipment addition to the spent fuel pool cleanup systems is necessary to maintain pool water quality for the augmented storage facility.

2.3 Radioactive Waste Treatment System

Each unit contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statements for Unit Nos. 1 and 2, dated February 1973 and June 1977, respectively. The proposed modifications will not result in any significant additional radwastes that will need to be processed. Therefore, there will be no changes in the waste treatment systems described in Section 3.0 of these Final Environmental Statements because of the proposed modifications.

3.0 NON-RADIOLOGICAL ENVIRONMENTAL IMPACTS OF PROPOSED ACTION

The non-radiological environmental impacts associated with the operations of ANO, as designed were considered in the FESs. The proposed modifications of SFPs will not cause any new non-radiological environmental impacts which were not previously considered based on the following:

1) The proposed modifications will alter only the spent fuel storage racks. It will not alter the external physical geometry of the SFP structures. In addition, construction of the new racks will be done offsite and transported to the facility. No unusual terrestrial effects are anticipated or considered likely.

2) Additional storage will not result in measurable increase in non-radiological chemical waste discharges to the receiving water. The licensee does not propose any change in chemical usage or change to the NPDES permit.

3) Additional SFP heat output will not cause measurable thermal effects to the receiving water. The increase in the heat load due to this modification is less than one tenth percent of the present SFP design heat load.

We conclude, based on the above evaluations, that the SFP modifications will not result in non-radiological environmental effects significantly greater or different from those already reviewed and analyzed in the FES for ANO-1 and ANO-2.

4.0 RADIOLOGICAL ENVIRONMENTAL IMPACTS OF PROPOSED ACTION

4.1 Introduction

The potential radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

During the storage of the spent fuel under water, both volatile and non-volatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90 are also predominantly nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFPs would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates, however, that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP.

During and after refueling, the SFP purification system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel is cooled in the SFP and the fuel clad temperature becomes relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the licensees and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the MO (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services (NFS) storage pool at West Valley, New York. Some spent fuel assemblies which had significant leakage while in operating reactors have been stored in these two pools. After storage in the onsite SFP, these fuel assemblies were later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from these fuel assemblies in the offsite storage facility.

4.2 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defective fuel. However, we have conservatively estimated that an additional 187.8 curies per year of Krypton-85 may be released when the ANO modified pools are completely filled. This increase would result in an additional total body dose to an individual at the site boundary of less than 0.001 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.003 person-rems/year. This is less than the natural fluctuations in the dose this population would receive from natural backgound radiation.

Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above the 150°F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FESs.

Most airborne releases of tritium and iodine result from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than the pool water. Therefore, even if there were a higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increased stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FESs. In addition, the station radiological effluent Technical Specifications limit the total releases of gaseous activity.

Based on the foregoing considerations, implementation of the proposed increased spent fuel storage capability will not result in significantly increased amounts of radioactivity being released to the atmosphere.

4.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity in the SFP water to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two resin beds (104 cubic feet wet) and two spent filter cartridges (20 cubic feet wet) per year from both units due to the increased operation of the spent fuel pool cleanup systems. The annual average volume of solid wastes shipped offsite for burial from a typical PWR with deep bed condensate demineralizer system is approximately 18,800 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 124 cubic feet (250 cubic feet solidified) per year from both units, the increase in total waste volume shipped from Arkansas Nuclear One would be less than 1% and would not have any significant additional environmental impact.

The present spent fuel racks to be removed from the SFPs because of the proposed modification are contaminated and may be disposed of as low level solid waste. We have estimated that approximately 14,000 cubic feet of solid radwaste will be removed from the plant because of the proposed modifications. Averaged over the lifetime of the plant this would increase the total waste volume shipped from the facility by less than 2%. This will not have any significant additional environmental impact.

4.4 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modifications. Since the SFP cooling and cleanup systems operate as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that the change in the quantity and activity of the floor cleanup water as a result of these modifications will be insignificant. The SFP demineralizer resin removes soluble radioactive materials from the SFP water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity of the SFP demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins. If any radioactive material is transferred from the spent resin to the sluice water, it will be removed by the liquid radwaste system for processing. After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

4.5 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks, and the installation of the high density racks, with respect to occupational radiation exposure. The occupational exposure for the operation is estimated by the licensee to be about 16 person-rems, based on the licensee's detailed breakdown of exposure to each individual performing specific jobs for each phase of the operation. This exposure is a small fraction of the total annual person-rem from occupational exposure for all plant operations.

We have estimated the increase in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of measured dose rates in the SFP area, and from radionuclide concentrations in the SFP water and from the SFP assemblies. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation dose to plant personnel. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational doses within the limits of 10 CFR 20, and ALARA.

5.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

5.1 Rack Module Assembly Drop Accident

The overhead cranes in the auxiliary building at ANO will be used for removing the existing rack modules and lowering the new modules into the pool. The licensee has stated in Section 8.1, Rack Modules Assembly Handling Considerations, of the November 5, 1982 submittal that "no loads exceeding 2000 lbs will be allowed over the fuel assemblies at any time." The Technical Specifications for ANO-1 and ANO-2 also prohibit the travel over fuel assemblies in the storage pool of loads in excess of 2000 lbs. Since the weight of a rack module is much greater than 2000 lbs, we conclude that the rack modules will not be carried over the fuel assemblies and that there is reasonable assurance that an accident impacting assemblies in the pool would not occur. Therefore, the assessment of the radiological consequences of a rack module assembly drop accident is not required.

5.2 Fuel Handling Accident

The maximum weight of loads which may be transported over spent fuel in the pool is limited by Technical Specifications to that of a single assembly (≈2000 lbs). The proposed spent fuel pool modification does not increase the radiological consequences of fuel handling accidents considered in the staff Safety Evaluation report of June 1973 (ANO-1) and November 1977 (ANO-2), since this accident would still result in, at most, the release of the gap activity of one fuel assembly due to the limitation on the available impact kinetic energy. In the evaluation of the cask drop accident, the licensee states in the November 5, 1982 submittal that the administrative procedures prevent the spent fuel cask from being moved over the spent fuel pool. The staff concludes that the proposed spent fuel pool modification does not affect the result of the cask drop accident considered in the staff's Safety Evaluation Reports.

5.3 Conclusion

Based upon the above evaluation, the staff concludes that the likelihood of a rack module assembly drop accident is sufficiently small - since the rack module assembly will not be allowed over the fuel at any time - that this accident need not be considered. Also, a fuel handling accident involving a dropped assembly or cask would not be expected to result in radionuclide releases leaking to offsite radiological consequences exceeding those of the fuel handling accident evaluated in the staff Safety Evaluation Reports of June 1973 (ANO-1) and November 1977 (ANO-2); that is, doses would be well within 10 CFR Part 100 values. We conclude therefore, that the proposed modifications are acceptable.

6.0 SUMMARY

The Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel concluded that the environmental impact of interim storage of spent fuel was negligible and the cost of the various alternatives reflect the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in SFP designs the FGEIS recommended licensing SFP expansion on a case-by-case basis. For ANO, expansion of the storage capacity of the SFPs does not significantly change the radiological impact evaluated in the FESs. As discussed in Section 4.5, the additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.001 mrem/year and 0.003 person-rems/year, respectively, and is less than the natural background radiation. Operation of ANO with additional spent fuel in the SFPs is not expected to increase the occupational radiation exposure by more than one percent of the total annual occupational exposure at ANO.

7.0 BASIS AND CONCLUSION FOR NOT PREPARING AN ENVIRONMENTAL IMPACT STATEMENT

We have reviewed the proposed modifications relative to the requirements set forth in 10 CFR Part 51 and the Council on Environmental Quality's Guidelines, 40 CFR 1500.6. We have determined, based on this assessment, that the proposed license amendments will not significantly affect the quality of the human environment. Therefore, the Commission has determined that an environmental impact statement need not be prepared and that, pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Date: April 15, 1983

Principal Contributors:
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8.0 REFERENCES

1. Arkansas Power and Light Company (AP&L) letter to U. S. Nuclear Regulatory Commission (USNRC) dated November 5, 1982 (OCAN118205).
2. AP&L letter to USNRC dated February 17, 1983 (OCAN028302).
3. AP&L letter to USNRC dated March 3, 1983 (OCAN028310).
4. AP&L letter to USNRC dated March 3, 1983 (OCAN028311).
5. AP&L letter to USNRC dated March 3, 1983 (OCAN038304).
6. AP&L letter to USNRC dated March 7, 1983 (OCAN038307).
7. AP&L letter to USNRC dated March 10, 1983 (OCAN038311).
8. AP&L letter to USNRC dated March 21, 1983 (OCAN038312).
9. AP&L letter to USNRC dated March 22, 1983 (OCAN038324).
10. AP&L letter to USNRC dated March 24, 1983 (OCAN038326).
11. AP&L letter to USNRC dated March 28, 1983 (OCAN038331).
12. AP&L letter to USNRC dated March 29, 1983 (OCAN048336).
13. AP&L letter to USNRC dated April 5, 1983 (OCAN048302).
14. AP&L letter to USNRC dated April 7, 1983 (OCAN048305).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-313 AND 50-368

ARKANSAS POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

AND NEGATIVE DECLARATION

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 76 and 43 to Facility Operating Licenses Nos. DPR-51 and NPF-6, issued to Arkansas Power and Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One, Units Nos. 1 and 2, respectively (ANO-1&2), located in Pope County, Arkansas. The amendments are effective as of the date of issuance.

The amendments allow an increase in the spent fuel storage capacity from 589 spaces to 968 spaces for ANO-1 and from 485 spaces to 988 spaces for ANO-2 through the use of high density storage racks.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Consideration of Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on December 22, 1982 (47 FR 57154).

No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. The Commission has prepared an

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environmental impact appraisal for this action and has concluded that an environmental impact statement for this particular action is not warranted because it will not significantly affect the quality of the human environment.

For further details with respect to this action, see (1) the application for amendments dated November 5, 1982, as supplemented February 17, 1983, and April 7, 1983, (2) Amendment No. 76 to License No. DPR-51 and Amendment No. 43 to License No. NPF-6, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Arkansas Tech University, Russellville, Arkansas. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 15th day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

NIRS

Nuclear Information and Resource Service
1346 Connecticut Avenue NW, 4th Floor, Washington, D.C. 20036 (202) 296-7552

March 6, 1984

Director
Office of Adminstration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

FREEDOM OF INFORMATION ACT REQUEST
FOIA-84-162
Rec'd 3-9-84

FREEDOM OF INFORMATION ACT REQUEST

To whom it may concern:

Pursuant to the Freedom of Information Act, 5 U.S.C. 522, as amended, the Nuclear Information and Resource Service requests the following documents regarding Portland General Electric's application to expand the storage capacity of its Spent Fuel Pool. Please consider "documents" to include reports, studies, test results, correspondence, memoranda, meeting notes, meeting minutes, working papers, graphs, charts, diagrams, notes and summaries of conversations and interviews, computer records, and any other forms of written communication, including internal NRC Staff memoranda. The documents are specifically requested from, but not limited to, the following offices of the NRC: Office of Nuclear Reactor Regulation (NRR); Office of Nuclear Regulatory Research (Research); the Operating Reactors Branches of the Division of Licensing; and the Office of the Executive Legal Director. In your response, please identify which documents correspond to which requests set out below.

Pursuant to this request, please provide all documents prepared or utilized by, in the possession of, or routed through the NRC related to:

1. NRC's review of Portland General Electric's application for a license amendment to increase the storage capacity of the Spent Fuel Pool including formal and informal correspondence and other communication both prior to and after receipt of the application, in particular regarding application of the "Sholly Amendment" provisions;

2. Technical reviews of PGE's letter of August 1, 1983 with the enclosed technical report designated PGE-1037, "Trojan Nuclear Plant Spent Fuel Storage Rack Replacement Report" or any other manner of written notation concerning the technical issues in this proposed amendment;

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3. Correspondence with members of Congress regarding this amendment and any NRC memoranda regarding such correspondence, lack or need thereof;

4. The scheduling of the no-significant-hazards-consideration determination and other reviews to be conducted by the Staff, including any reviews, comments or responses to PGE's letter of February 6, 1984 from PGE to James R. Miller of the NRC Staff and subsequent correspondence;

In our opinion, it is appropriate in this case for you to waive copying and search charges, pursuant to 5 U.S.C. 552(a)(4)(A) "because furnishing the information can be considered as primarily benefiting the general public." The Nuclear Information and Resource Service is a non-profit organization serving local organizations concerned about nuclear power and providing information to the general public.

Please note that it is expected that this request will be processed within the ten-day period allowed by law. Additionally, it is asked that this search be conducted without the extension of time usually required by the FOIA Staff and generally granted by this requestor.

Sincerely,

Nina Bell

Nina Bell
Nuclear Safety Analyst

cc: File