June 23, 1983

POLICY ISSUE

(NEGATIVE CONSENT)

SECY-83-249

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 on July 11, 1983, unless notified to the contrary, the staff proposes to issue 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 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. A copy of the licensee's submittal is enclosed.

The staff has reviewed a detailed NSHC determination included in the licensee's submittal and has 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, thermal-hydraulic, or 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.

The Commission is being advised of this action in view of the guidance provided with regard to spent fuel pool reracking in the publication of the Interim Final Rule as part of 10 CFR 50. This guidance provided, in part, that NSHC findings for reracking applications would be made on a case-by-case basis (48 FR 14869). Moreover, the legislative history of P.L. 97-415 and continuing Congressional interest in the subject of spent fuel pool rerackings, dictates that the Commission be aware of the staff's proposed action.

As you know, we have issued a Federal Register notice regarding our proposed determination that the TMI-1 steam generator repair issue does not involve significant hazards considerations. We anticipate at least one request for hearing relative to this matter and, therefore, anticipate the need for a final significant hazards determination. We plan to discuss this issue with the Commission prior to making this determination.

William J. Dircks
Executive Director for Operations

Enclosures:
As Stated

Contact:
J. F. Suermann
X27471

SECY NOTE: In the absence of instructions to the contrary, SECY will notify the staff on Friday, July 8, 1983 that the Commission, by negative consent, assents to the action proposed in this paper.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

DOCKET NO. 50-287

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. DPR-55, issued to Duke Power Company (the licensee), for operation of the Oconee Nuclear Station, Unit No. 3 (the facility) located in Oconee County, South Carolina.

In accordance with the licensee's application for amendment dated March 10, 1983, the amendment would permit the expansion of the spent fuel storage capacity for Oconee Unit No. 3. This expansion would be accomplished by reracking the existing spent fuel storage pool with neutron absorbing (poison) spent fuel racks. Reracking the spent fuel pool would increase the Oconee Unit No. 3 pool storage capacity from 474 to 825 spaces.

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 of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

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The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). Spent fuel pool reracking was specifically excluded from either set of examples because "[for reracking]...a significant hazards consideration finding is a technical matter which has been assigned to the Commission..." and the Commission "...will make a finding...for each reracking application, on a case-by-case basis..." (48 FR 14869). In this instance, the licensee's submittal of March 10, 1983 (hereafter referred to as the submittal) included a discussion of the proposed action with respect to the no significant hazards consideration. This discussion has been reviewed and the Commission finds it acceptable. Each of the three standards is discussed below.

First Standard

The analysis of the proposed reracking has been accomplished using current NRC Staff accepted Codes and Standards as specified in Section 2.1.2 of Attachment 2 of the submittal. The results of the analysis meet the specified acceptance criteria set forth in these standards. In addition, Duke has reviewed NRC Staff Safety Evaluation Reports for prior PWR rerackings involving poison racks to ensure that there are no identified concerns not fully addressed in their submittal.

From its analyses and SER reviews Duke has identified the following potential accident scenarios: (1) spent fuel cask drop; (2) loss of spent fuel pool forced cooling; (3) seismic event; (4) spent fuel assembly drop; and (5) construction accident. The probability of any of the first four accidents is not affected by the racks themselves; thus, reracking cannot increase the probability of these accidents. As for the construction

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accident, the proposed Oconee 3 pool reracking will not involve an increase in probability of any previously evaluated construction accident as accepted construction standards and procedures will be employed as described in Sections 4.0 and 6.1 of Attachment 2 of the submittal. Since there will be no fuel assemblies in the fuel pool during rack installation, the probability of some types of postulated construction accidents has actually decreased.

The consequences of the (1) spent fuel cask drop accident have been evaluated as described in Section 6.2 of Attachment 2 of the submittal. By limiting the age of fuel stored in the first 31 rows to not less than 70 days prior to any cask movement, Duke indicates that the consequences of this type accident would be less than with the present racks as described in the Oconee FSAR Section 15.11.2.2. Thus, the consequences of this type accident would not be significantly increased from previous accident analyses.

The consequences of the (2) loss of spent fuel pool forced cooling accident have been evaluated and are described in Section 6.3 of Attachment 2 of the submittal. As indicated by Duke in Tables 6.3-1 and 6.3-2, there is ample time to effect repairs to the cooling system or to establish a makeup flow, and since the required makeup flow is less than the 70 gpm rate accepted by the NRC Staff for the Oconee 1 and 2 pool, the consequences of this type accident would not be significantly increased from previously evaluated accidents by this proposed reracking.

The consequences of a (3) seismic event have been evaluated and are described in Section 2.3.1 of Attachment 2 of the submittal. The racks were evaluated against the appropriate NRC Standard described in Section 2.1.2. Duke indicates that the results of the seismic and structural analysis show

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that the proposed racks meet all of the NRC structural acceptance criteria and are consiste it with results found acceptable by the NRC Staff in all previous poison r k SERs including Oconee 1 and 2. Thus, the consequences of seismic events would not significantly increase from previously evaluated seismic events.

The consequences of a (4) spent fuel assembly drop accident are described in Section 2.3.1.5 of Attachment 2 of the submittal. The radiological consequences of this type accident are bounded by the cask drop accident and Duke indicates that K_{eff} is shown to be always less than the NRC acceptance criteria of 0.95 and not significantly different from the margin to criticality found in the December 22, 1975 SER for the previous Oconee 3 rerack. Thus, the consequences of this type accident would not be significantly increased from previously evaluated spent fuel assembly drop accidents.

The consequences of a (5) construction accident are described in Section 6.1 of Attachment 2 of the submittal. Since there will be no fuel assemblies in the fuel pool during rack installation, there would be no radiological consequences of any construction accident. Thus, using accepted construction practices as described in Section 4.0 of Attachment 2 of the submittal the consequences of a construction accident would be less than construction accidents previously evaluated by the NRC Staff.

Based on the information provided with the application, the proposed Oconee 3 spent fuel pool rerack would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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Second Standard

Duke has evaluated the proposed reracking 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 Plans, and appropriate Industry Codes and Standards as described in Section 2.1.2 of Attachment 2 of the submittal. In addition, Duke has reviewed previous NRC Safety Evaluation Reports for poison rerack applications. In Duke's analysis and review of NRC evaluations and Industry Standards and Codes, Duke finds that the proposed reracking does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated including those on the Oconee 3 Docket.

Third Standard

The issue of margin of safety when applied to a reracking modification will need to address the following areas (as established by the NRC Staff Safety Evaluation review process):

1. Nuclear criticality considerations
2. Thermal-hydraulic considerations
3. 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 spent fuel pool is to be less than or equal to 0.95, including all uncertainties, under all conditions. For the proposed modification, the criticality analysis, as discussed in Section 2.3.2 of Attachment 2 of the submittal is exactly the same as that which was approved by the NRC Staff (SER issued December 24, 1980) for the Unit 1 and 2 shared pool reracking modification. The exact same codes, techniques, and assumptions were made. All aspects of the bases of the SER conclusions are covered in the identical manner.

The methods utilized in the analysis conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 5.1.12; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety," NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage;" and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The results of Duke's analysis indicate that K_{eff} is always less than 0.95 including uncertainties at a 95/95 probability/confidence level. Thus meeting the acceptance criteria for criticality, the proposed rerack 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 2.3.3 of Attachment 2 of the submittal. Results of these analyses by Duke 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

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storage cells. However, the proposed reracking will allow an increase in the heat load in the Oconee 3 spent fuel pool. The evaluation in Section 3 of Attachment 2 of the submittal shows that a third spent fuel cooling train will be added prior to putting more than the currently authorized 474 Fuel Assemblies in the spent fuel pool. The addition of the third cooling train is intended to ensure that the pool temperature margins of safety of 150°F and 205°F described in Section 9.1.3 of the Oconee FSAR are maintained. Thus, there would be no significant reduction in the margin of safety from a thermal-hydraulic standpoint or from a spent fuel cooling standpoint.

The mechanical, material, and structural considerations of the proposed rerack are described in Sections 2.1, 2.2, and 2.3 of Attachment 2 of the submittal. As described by Duke in Section 2.1, the racks are designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979. The racks are designed to Seismic Category 1 requirements and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support Structures. In addition, the racks are designed to withstand the loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane. Duke indicates that the materials utilized are described in Sections 2.2 and 2.3.4 and are compatible with the spent fuel pool and the spent fuel assemblies. The structural considerations of the racks are described in Section 2.3 and show that the margin of safety against tilting is greater than 100, that the racks do not impact each other nor impact the pool walls, and that sufficient clearance is provided to prevent the racks from sliding into pool floor obstructions. Thus, the margin of safety would not be significantly reduced by the proposed rerack.

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Because the submittal by the licensee appears 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's staff proposes to determine that the application does not involve a significant hazard 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.

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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.

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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 detemination 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.

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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 John F. Stolz: 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. Michael McGarry, III, DeBevoise & Liberman, 1200 17th Street, N.W., Washington, D.C. 20036, 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).

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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 Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina.

Dated at Bethesda, Maryland, this day of

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

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ENCLOSURE 1

ENCLOSURE 2

Enclosure 2

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

March 10, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4

Subject: Oconee Nuclear Station
Docket No. 50-287

Dear Sir:

Pursuant to 10 CFR §50.90, please find attached a proposed amendment to the Oconee Nuclear Station Facility Operating License and proposed revision to the Oconee Technical Specifications. Attachment 1 contains the proposed Technical Specification revision which concerns expansion of the storage capacity of the Oconee Unit 3 spent fuel pool from 474 to 825 spaces. The proposed expansion is to be achieved by reracking the spent fuel pool with poison racks, the safety and environmental implications of which are addressed in Attachment 2.

By letter dated December 24, 1980, the Commission approved the Oconee Units 1 and 2 shared spent fuel pool reracking with neutron absorbing (poison) spent fuel racks. For the propsed Oconee Unit 3 spent fuel pool reracking, Duke Power Company ("Duke") has chosen the same vendor to provide essentially identical racks. The analysis provided in Attachment 2 addresses all of the areas addressed in the December 24, 1980 evaluation in the same manner. The results of this analysis indicate that there are no outstanding safety issues in this proposal.

Duke's current schedule calls for reracking to begin in September 1983, and to be completed by March 1984. Utilizing this schedule the reracking can be accomplished with all spent fuel removed from the Unit 3 pool and stored in the Oconee 1 and 2 pool. This would make the reracking operation much simpler and safer.

Duke submits that the activities associated with the amendment do not constitute a significant hazard to the public health and safety or to the environment. Thus, Duke respectfully requests that the NRC Staff process this application pursuant to 10 CFR §§50.91 and 50.92 as they pertain to applications involving no significant hazards consideration. 1/ In accordance with 10 CFR §50.91(a), attached hereto (Attachment 3) is an analysis of the proposed action in light of the standards contained in §50.92(b) regarding the issue of no significant hazards consideration. Further, in accordance with 10 CFR §50.91, Duke is forwarding a copy of this application to the South Carolina Department of Health & Environmental Control for review and, as appropriate, subsequent consultation with the Staff.

This proposal is considered to consist of one Class III license amendment. Accordingly, please find attached a check in the amount of $4,000.

Very truly yours,

s/Hal B. Tucker
Hal B. Tucker

JFN/php
Attachment

cc: Mr. James P. O'Reilly
Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

Mr. Eben L. Conner, Jr.
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Hayward Shealey, Chief
Bureau of Radiological Health
S. C. Department of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. J. C. Bryant
NRC Resident Inspector
Oconee Nuclear Station

1/ The reference to 10 CFR §§50.91 and 92 is to the proposed amendments set forth in SECY-83-16, as revised (Duke notes that the standards set forth in SECY-83-16 are indeed the standards that have been used by the NRC Staff for many years). Duke understands that these regulations will be approved by the Commission within the next several days. Duke also understands that such amendment will not become effective until 30 days after publication in the Federal Register. Due to the need to timely complete the requested reracking, Duke is submitting this application to the NRC in advance of the effective date of the regulations. Duke requests that such regulations, once approved by the Commission, be made applicable to the instant spent fuel pool modification. Specifically, Duke requests that an appropriate notice be timely published in the Federal Register, that NRC Staff review commence, and that a no significant safety hazards consideration finding be made, if necessary.

HAL B. TUCKER, being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

s/Hal B. Tucker
Hal B. Tucker, Vice President

Subscribed and sworn to before me this 10th day of March, 1983.

s/Sue C. Sherrill
Notary Public

My Commission Expires:

September 20, 1984

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 1

Proposed Technical Specification Revision

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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.

3.8.10 The reactor building purge system, including the radiation monitor, RIA-45, which initiates purge isolation, shall be tested and verified to be operable immediately prior to refueling operations.

3.8.11 Irradiated fuel shall not be moved from the reactor until the unit has been subcritical for at least 72 hours.

3.8.12 Two trains of spent fuel pool ventilation shall be operable with the following exceptions:

a. With one train of spent fuel pool ventilation inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the operable spent fuel pool ventilation train is in operation and discharging through the Reactor Building purge filters.

b. With no spent fuel pool ventilation filter operable, suspend all operations involving movement of fuel within the storage pool or crane operations with loads over the storage pool until at least one train of spent fuel pool ventilation is restored to operable status.

c. This specification does not apply during reracking operations with no fuel in the spent fuel pool.

3.8.13 a. Prior to spent fuel cask movement in the Unit 1 and 2 spent fuel pool, spent fuel stored in the first 36 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 55 days.

b. Prior to spent fuel cask movement in the Unit 3 spent fuel pool, spent fuel stored in the first 31 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 70 days.

3.8.14 No suspended loads of more than 3000 lb_m shall be transported over spent fuel stored in either spent fuel pool.

3.8.15 a. No fuel which has an enrichment greater than 4.0 weight percent U^{235} (53 grams of U^{235} per axial centimeter of fuel assembly) will be stored in the spent fuel pool for Unit 3.

b. No fuel which has an enrichment greater than 4.3 weight percent U^{235} (57 grams of U^{235} per axial centimeter of fuel assembly) will be stored in the spent fuel pool for Units 1 and 2.

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 9.1.4 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 low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above 1835 ppm. Although this 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 0.90. 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 of the Reactor Building purge isolation 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.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.(3)

The off-site doses for the fuel handling accident are within the guidelines of 10 CFR 100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.8.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

REFERENCES

(1) FSAR, Section 9.1.4
(2) FSAR, Section 15.11.1
(3) FSAR, Section 15.11.2.1

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Specification

5.4.1 New Fuel Storage

5.4.1.1 New fuel will normally be stored in the spent fuel pool serving the respective unit.

In the spent fuel pool serving Units 1 and 2, the fuel assemblies are stored in racks in parallel rows, having a nominal center-to-center distance of 10.65 inches in both directions. This spacing is sufficient to maintain $K_{eff} \leq 0.95$ when flooded with unborated water, based on fuel with an enrichment of 4.3 weight percent U^{235}.

In the spent fuel pool serving Unit 3, the fuel assemblies are stored in racks in parallel rows, having a nominal center-to-center distance of 10.60 inches in both directions. This spacing is sufficient to maintain a $K_{eff} \leq 0.95$ when flooded with unborated water, based on fuel with an enrichment of 4.0 weight percent U^{235}.

5.4.1.2 New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center-to-center distance of 2' 1-3/4". One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.

5.4.1.3 New fuel may also be stored in shipping containers.

5.4.2 Spent Fuel Storage

5.4.2.1 Irradiated fuel assemblies will be stored, prior to off-site shipment, in a stainless steel lined spent fuel pool.

The spent fuel pool serving Units 1 and 2 is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program.

Provisions are made in the Units 1, 2 spent fuel pool to accommodate up to 1312 fuel assemblies and in the Unit 3 spent fuel pool up to 825 fuel assemblies.

5.4.2.2 Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

5.4.3 Whenever there is fuel in the pool, the spent fuel pool is filled with water borated to the concentration that is used in the reactor cavity and fuel transfer canal during refueling operations.

5.4.4 The spent fuel pool and fuel transfer canal racks are designed for an earthquake force of 0.1g ground motion.

REFERENCES

FSAR, Section 9.1.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

Attachment 2

Unit 3 Spent Fuel Pool
Poison Rerack Licensing Submittal

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1.0 INTRODUCTION

The Oconee Nuclear Station was designed and constructed with two spent fuel storage pools--one associated with Units 1 and 2 and one with Unit 3. The design was such that the pools would be capable of storing 1 2/3 and 1 1/3 cores, respectively. Both the Oconee Nuclear Station Final Safety Analysis Report and Technical Specifications address the adequacy of this to "accommodate a full core of irradiated fuel assemblies in addition to concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program" (References 1 and 2). The actual designed capacity for each pool was 336 and 216 locations.

In 1975 it was deemed prudent to increase the storage capacity at the Oconee site. The Unit 1 and 2 pool contained spent fuel from the initial Unit 1 refueling in 1974. The Unit 3 pool was empty of any spent fuel. Thus, it was decided to increase the capacity of the Unit 3 pool. A request to amend the Unit 3 Operating License, DPR-55, was submitted on September 12, 1975 and was approved, as Amendment No. 17, on December 22, 1975. The completed modification increased its capacity to 474 locations (including failed fuel). It was considered at that time that the resulting combined on-site capacity (810 locations) would be sufficient to store spent fuel until such time as shipment to the Allied General Nuclear Services reprocessing plant could begin. The modifications of the Unit 1 and 2 pool in this time frame would have had to have been done "wet" (with spent fuel present in pool). Such operations were considered to exceed state-of-the-art technical capabilities at that time.

On April 17, 1977, President Carter issued a policy statement on commercial reprocessing of spent nuclear fuel which effectively eliminated reprocessing as part of the nuclear fuel cycle, at least in the near future. On October 18, 1977, the Department of Energy accepted ultimate responsibility for storing spent nuclear fuel. On December 23, 1977, the GESMO proceedings were deferred indefinitely. The combined effect of this national policy was to leave operating nuclear plants, like Oconee, without a repository for the spent fuel previously generated or being generated, other than expanded storage provided by the owner/operator. Thus, Duke Power has been forced to rerack the Oconee pools to provide further storage capacity.

By letter dated February 2, 1979, Duke requested approval of expanding the capacity of the Unit 1-2 pool utilizing high-capacity non-poison racks. The expansion of the Unit 1 and 2 pool capacity as approved, allowed the storage of up to 750 assemblies in that pool and 1224 on-site (including "failed-fuel" locations).

By letter dated July 25, 1980, as supplemented by seven other submittals, Duke requested approval for using Westinghouse designed/constructed poison racks in the Oconee 1 and 2 pool. By letter dated December 24, 1980, the NRC issued Amendments 90, 90, and 87 which authorized the rerack and thus increased the Unit 1-2 pool fuel storage capacity from 750 to a maximum of 1312 fuel assemblies.

The current fuel storage capacity at Oconee thus consists of 1312 storage spaces in the Oconee 1-2 shared pool and 474 spaces in the Oconee 3 pool. With this submittal Duke Power is requesting approval for using Westinghouse designed/constructed poison racks to increase the Oconee 3 storage capacity to 825 spaces. This modification would extend the Oconee fuel storage capacity from the current September 1988 date to October 1991. With the proposed rerack the full core reserve capacity would be extended from January 1988 to March 1990.

The increase in Oconee 3 storage capacity would be accomplished by replacing the existing 14.09 inch center to center high density racks with 10.60 inch center to center neutron absorbing racks. These racks would be similar to those utilized in the Oconee 1 and 2 storage pool and thus are of proven design and installation.

The following chapters are provided with intent to provide information necessary for review and approval of the request for amendment to the Oconee Nuclear Station Technical Specifications (Attachment 1). It is considered that the modification is not inimical to the health and safety of personnel or the general public and that it represents an environmentally acceptable alternative which meets the requirements of NEPA and the guidance provided by the Commission on such applications. The modification involves no significant hazards and will help Oconee meet the intent of the Nuclear Waste Policy Act (PL 97-425) for on-site storage.

References

1. Oconee Nuclear Station Technical Specification Section 5.4.2.1.
2. Oconee Nuclear Station Final Safety Analysis Report Section 9.1.4.1.3.
3. "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."
4. September 16, 1975 Federal Register Notice (FR-42801)

2.0 RACK DESIGN

2.1 DESIGN BASES

The function of the spent fuel storage racks is to provide for storage of spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings.

A list of design criteria is given below:

1. The racks are designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 and revised January 18, 1979.

2. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor, K_{eff}, in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions as described in Section 2.3.2.

3. The racks are designed to allow coolant flow such that boiling in the water channels between the fuel assemblies in the rack does not occur. Maximum fuel cladding temperatures are calculated for various pool cooling conditions as described in Section 2.3.3.

4. The racks are designed to Seismic Category I requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support structures. The structural evaluation and seismic analyses are performed using the specified loads and load combinations in Section 2.1.1.

5. The racks are designed to withstand loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane.

6. Each storage position in the racks is designed to support and guide the fuel assembly in a manner that will minimize the possibility of application of excessive lateral, axial and bending loads to fuel assemblies during fuel assembly handling and storage.

7. The racks are designed to preclude the insertion of a fuel assembly in other than design locations.

8. The materials used in construction of the racks are compatible with the storage pool environment and do not contaminate the fuel assemblies.

2.1.1 Specified Loads and Definitions

The following are load combinations specified for racks:

Elastic Analysis	Acceptance Limits
(1) D + L	Normal Limits of NF 3231.1a
(2) D + L + E	Normal Limits of NF 3231.1a
(3) D + L + T_o	Lesser of 2 S_y or S_u Stress Range
(4) D + L + T_o + E	Lesser of 2 S_y or S_u Stress Range
(5) D + L + T_a + E	Lesser of 2 S_y or S_u Stress Range
(6) D + L + T_a + E'	Faulted Condition Limits of NF 3231.1c

Definitions:

D - Dead loads or their related internal moments and forces including any permanent equipment and hydrostatic loads.

L - Live loads or their related internal moments and forces including any movable equipment loads.

T_o - Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

T_a - Thermal effects and loads resulting from the highest temperature associated with the postulated abnormal design condition.

E - Loads generated by the operating basis earthquake.

E' - Loads generated by the safe shutdown earthquake.

Analyses were performed to verify the acceptability of the critical load components and paths under the load combinations given above.

2.1.2 Applicable Codes and Standards

"NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979.

NRC Regulatory Guides

R.G. 1.13 Spent Fuel Storage Facility Design Basis

R.G. 1.29 Seismic Design Classifications

R.G. 1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants

R.G. 1.61 Damping Values for Seismic Design of Nuclear Power Plants

R.G. 1.92 Combining Modal Responses and Spatial Components in Seismic Response Analysis

R.G. 1.124 Service Limits and Loading Combinations for Class I Linear-Type Component Supports

NRC Standard Review Plans

SRP 3.7 Seismic Design

SRP 3.8.4 Other Category I Structures

SRP 9.1.2 Spent Fuel Storage

SRP 9.1.3 Spent Fuel Pool Cooling and Cleanup System

Industry Codes and Standards

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

American National Standards Institute, N210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

American National Standards Institute, N16.1-1975, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

2.2 DESIGN DESCRIPTION

The spent fuel storage rack is composed of individual storage cells made of stainless steel. Each cell has a lead-in opening which is symmetrical and is blended smooth. This opening precludes insertion of the fuel assemblies in other than the prescribed locations. These racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. The cells within a module are interconnected by grid assemblies to form an integral structure as shown in Figure 2-1. Each rack module is provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls. The following information applies to Oconee Unit 3 spent fuel storage racks.

Number of Cells	822 plus storage locations for 3 failed fuel containers
Number Rack Arrays	7 - 8 x 10 2 - 8 x 12 1 - 8 x 10 w/3 container locations
Poison Material	Boraflex 0.03 gm $^{10}B/cm^2$ Vented to pool environment
Center-to-Center Spacing	10.60 in.
Type of fuel	B&W 15 x 15, 4.0 weight percent enrichment (maximum)
Rack Assembly Dimension and Weights	8 x 10 - 85.5 x 107 x 172 - 20,200 lbs. 8 x 12 - 85.5 x 128 x 172 - 24,000 lbs.

The pool outline and rack arrangement is shown in Figure 2-2.

2.2.1 Design Loads

D - Weight of 8 x 10 rack dry . . . . . 20,000 lbs.
Weight of 8 x 12 rack dry . . . . . 24,000 lbs.

L - Live loads are negligible since the fuel assemblies are lowered very slowly into the cells.

T_o - Service expansion temperature range $\Delta T = 20°F$

T_a - A postulated design condition that would cause T_a is failure of the spent fuel pool cooling system. The water will gradually heat up and boiling could theoretically occur; however, since this process is slow, it is predicted to remain in the T_o ΔT range.

E - OBE loads.

E' - SSE loads.

2.3 DESIGN EVALUATION

Evaluations and analyses were performed in the following areas to verify the ability of the rack design to perform its required functions.

1. Structural and Seismic

2. Nuclear Criticality

3. Thermal-Hydraulic

4. Poison Material

2.3.1 Structural And Seismic

The purpose of the structural analysis is to analyze the critical components/load paths under various loading conditions. The structural analysis also determines the margin of safety against overturning due to loads from an SSE. The racks rest freely on the pool floor and are evaluated to ensure that under various loading conditions they do not impact each other, nor do they impact the pool walls. Sufficient clearance is also provided to prevent the racks from sliding into pool floor obstructions. Figure 2-3 shows the general arrangement of a typical fuel rack assembly and its pool floor leveling pad.

2.3.1.1 Component Description

The complete fuel rack assembly is divided into three major sections for stress analysis purposes:

1. Rack support assembly

2. Lower and upper grid assembly

3. Cell assembly

The following paragraphs describe each assembly:

Rack Support Assembly

The Rack Support Assembly consists of the Support Block, Leveling Pad Assembly and Standoff Blocks as appropriate.

The top of the support block is welded to the base plate. The leveling pad assemblies transmit the loads to the pool floor and provide a sliding contact. There are ten leveling pad assemblies for each 8 x 10 and twelve for each 8 x 12 rack assembly. The leveling pad screw permits the leveling adjustment of the rack. The major components of the leveling pad assembly are the leveling pad and the leveling pad screw.

Lower and Upper Grid Assembly

The lower grid attaches the cell assembly to the base plate. The lower grid consists of box-beam members, the side plates and the base plate. The cell assembly at bottom is welded to the lower grid through integral cell wall dimples. The upper grid consists of the box-beam members and the side plates. The cell assembly at the top is welded to the upper grid through integral cell wall dimples. The upper and lower grid assembly maintains the precise center-line to center-line spacing between the cells and provides the structural connections between the cells to form a fuel rack assembly.

Cell Assembly

The major components of the cell assembly are the fuel assembly cell, the Boraflex (neutron absorbing) material, and the wrapper.

The ID of the cell is 9.085 with a 0.075 inch wall. The upper end of the cell has a funnel shape flare for easy insertion of the fuel assembly. The wrapper is attached to the outside of the cell through spot welding along the length of the wrapper. Thus, the wrapper surrounds the Boraflex material, and also provides for venting to the pool environment. Dimples are formed in the upper and lower cell walls to position the cell within grid assembly openings and to provide for a structural weld connection between the cell and the grid assembly.

2.3.1.2 Seismic Analysis Models

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the structure. The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model shown in Figure 2-4(A). The second phase is a response spectrum analysis of a detail rack assembly finite element model shown in Figure 2-4(B). The damping values used in the seismic analysis are two percent damping for OBE and four percent damping for SSE as specified in NRC Regulatory Guide 1.61.

The simplified nonlinear finite element model is used to determine the fuel rack response for full, partially filled, and empty fuel assembly loading conditions. This nonlinear model has the structural characteristics of an individual cell within a submerged rack assembly. The nonlinearities of the fuel rack assembly which are accounted for in the model are due to changes in the gap between the fuel cell and the fuel assembly, the boundary conditions of the fuel rack support locations and energy losses at the support locations.

The fuel assembly to cell impact loads, support pad lift off, rack sliding, and overall rack response are obtained from the nonlinear time history model. In determining the maximum fuel rack response, the response value for each item of interest is searched for maximum values.

The detail model is a three-dimensional finite element representative of a rack assembly consisting of discrete three-dimensional beams interconnected at a finite number of nodal points.

The results of the single cell nonlinear time history model are incorporated in the detail model. Since the detail model does not account for the nonlinear effect of a fuel assembly impacting the cell and the support pad movements, the internal loads and stresses for the rack assembly obtained from this model are corrected by load correction factors. The load correction factors are derived from the single cell nonlinear model results and are applied to the components in the structural analysis. The responses of the model from accelerations in three directions are combined by the SRSS method in the structural analysis. The loads in four major components (support pad assembly, bottom grid, top grid, and fuel cell) are examined, and the maximum loaded section of each of these components was found. These maximum loads from the detail model are used in the structural analysis to obtain the stresses within the rack assembly.

2.3.1.3 Loads and Load Combinations for Structural Analysis

The loads and load combinations to be considered are those given in NRC Standard Review Plan, Section 3.8.4-II.3. The thermal loads due to rack expansion relative to the pool floor are negligible since the support pads are not structurally restrained in the lateral direction. The major seismic loads are produced by the operational basis earthquake (OBE) and safe shutdown earthquake (SSE) events.

It is noted from the seismic analysis that the magnitude of stresses vary considerably from one geometrical location to the other in the model. Consequently, the maximum loaded cell assembly, grid assembly and the leveling pad assembly are analyzed. Such an analysis envelopes the other areas of the rack assembly.

The margins of safety for the multi-direction seismic event are produced by combining x-direction and y-direction loads by the SRSS method.

The loads used in the seismic analysis are corrected by load correction factors obtained from the nonlinear analysis.

2.3.1.4 Fuel Handling Crane Uplift Analysis

The objective of this analysis is to ensure that the rack can withstand the maximum uplift load of 3000 pounds of the fuel handling crane without violating the criticality acceptance criteria.

Two accident loading conditions are postulated. The first condition assumes that the uplift load is applied to a fuel cell. The second condition assumes that the load is applied to the top grid. Calculations show that for either condition, the resulting stresses are within acceptable stress limits. There is no change in rack geometry and the criticality acceptance criteria are not violated.

2.3.1.5 Fuel Assembly Drop Accident Analysis

The objectives of this analysis are to ensure that, in the unlikely event of dropping a fuel assembly, accidental deformation to the rack will not cause the criticality acceptance criteria to be violated, and the spent fuel pool liner will not be perforated.

Two accident conditions are postulated. The first accident condition assumes that the weight of a fuel assembly, control rod assembly and handling mechanism of 3000 pounds impacts the top end fitting of a stored fuel assembly from a drop height of 6 feet. Calculations show that the impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cell funnels, the section of cell above the upper grid and the rack base plate/lower grid assembly. If in the unlikely event that two adjacent cells are crushed together for their full length, criticality calculations show that $K_{eff} < 0.95$. Under these faulted conditions, credit is taken for dissolved boron in the water, and the criticality acceptance criteria is not violated for the Oconee poison spent fuel racks. The pool liner is not perforated. A radiological evaluation is provided in Section 6.2.

The second accident condition assumes that the fuel assembly (3000 lbs.) falls straight through an empty cell, and impacts the rack base plate from a drop height of 234 inches. The results of this analysis show that the impact energy is absorbed by the fuel assembly and the rack base plate. The spent fuel pool liner is not perforated and the margin of safety is positive. Criticality calculations show that $K_{eff} < 0.95$ and the criticality acceptance criteria is not violated for the Oconee poison spent fuel racks.

In both these accident conditions, the criticality acceptance criteria is not violated and the spent fuel pool liner is not perforated.

2.3.1.6 Fuel Rack Sliding and Overturning Analysis

Consistent with the criteria of "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," the racks are evaluated for overturning and sliding displacement due to earthquake conditions.

The nonlinear model described in paragraph 2.3.1.2 is used in this evaluation to account for fuel-to-rack impact loading, hydrodynamic forces, and the nonlinearity of sliding friction interfaces.

The horizontal resistive force at the interface between the rack module and pool floor is produced by friction. A low coefficient of friction ($\mu = 0.2$) produces maximum rack horizontal displacement or sliding while a high value ($\mu = 0.8$) produces maximum rack horizontal overturning force.

The fuel rack nonlinear time history analysis shows that the fuel rack slides a minimal distance (< .200 inches). This distance is less than the rack-to-rack, rack-to-floor obstruction, or rack-to-wall clearances; thus, impact between adjacent rack modules, rack module and floor obstructions, and rack module and pool wall is prevented. Also, the factor of safety against tilting is > 100 which is well within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

2.3.1.7 Structural Acceptance Criteria

The fuel racks are analyzed for the normal and faulted load combinations of Section 2.1.1 in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The major normal and upset condition loads are produced by the operational basis earthquakes (OBE). The thermal stresses due to rack expansion relative to the pool floor are negligible since the support pads are not structurally restrained in the lateral direction.

The faulted condition loads are produced by the safe shutdown earthquakes (SSE) and a postulated fuel assembly drop accident.

The computed stresses are below the allowable stresses as required by the ASME B&PV Code, Section III, Subsection NF.

In summary, the results of the seismic and structural analysis show that the Oconee spent fuel storage racks meet all the structural acceptance criteria adequately.

2.3.2 <u>Nuclear Criticality</u>

2.3.2.1 Neutron Multiplication Factor

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI N210-1976 and in "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The following are the conditions that are assumed in meeting this design basis:

2.3.2.2 Normal Storage

1. The fuel assembly contains the highest enrichment authorized without any control rods or any noncontained burnable poison and is at its most reactive point in life. The enrichment of the fuel assembly is 4.0 w/o U-235 with no depletion or fission product buildup.

2. The moderator is pure water at the temperature within the design limits of the pool which yields the largest reactivity. A conservative value of 1.0 gm/cm^3 is used for the density of water. No dissolved boron is included in the water.

3. The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design. The nominal case calculation is infinite in lateral and axial extent. However, poison plates are not necessary on the periphery of the modular array and between widely spaced modules because calculations show that this finite array is less reactive than the nominal case infinite array. Therefore, the nominal case of an infinite array of poison cells is a conservative assumption.

4. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate values. The items included in the analysis are:

   - Poison pocket thickness
   - Stainless steel thickness
   - Can ID
   - Center-to-center spacing
   - Can bowing

   The calculational method uncertainty and bias is discussed in Section 2.3.2.4.

5. Credit is taken for the neutron absorption in full length structural materials and in solid materials added specifically for neutron absorption. A minimum poison loading is assumed in the poison plates and B_4C particle self shielding is included as a bias in the reactivity calculation.

2.3.2.3 Postulated Accidents

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the dropped assembly has more than eight inches of water separating it from the active fuel height of stored assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity. Therefore, for accident conditions, the double contingency principle of ANS N16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The presence of approximately 2000 ppm boron in the pool water will decrease reactivity by about 30 percent Δk. In perspective, this is more negative reactivity than is present in the poison plates (25 percent Δk), so K_{eff} for the rack would be less than 0.95 even if the poison plates were not present. Thus, for postulated accidents, should there be a reactivity increase, K_{eff} would be less than or equal to 0.95 due to the combined effects of the dissolved boron and the poison plates.

The "optimum moderation" accident is not a problem in spent fuel storage racks because possible water densities are too low ($\leq$ 0.01 gm/cm^3) to yield K_{eff} values higher than for full density water and the rack design prevents the preferential reduction of water density between the cells of a rack (e.g., boiling between cells). Further, the presence of poison plates removes the conditions necessary for "optimum moderation" so that K_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 to 0.0 gm/cm^3 in poison rack design.

2.3.2.4 Criticality Analysis

The calculation method and cross-section values are verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which insures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes[1,2] for cross-section generation and KENO IV[3] for reactivity determination.

The 218 energy group cross-section library[1] that is the common starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-IV data. The NITAWL program[2] includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM program[2] which is a one-dimensional S_N transport theory code. These multigroup cross-section sets are then used as input to KENO IV[3] which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 27 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated, oxide fuel arrays separated by various materials (Boral, steel, water) that simulate LWR fuel shipping and storage conditions[4,5] to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials[6] (Plexiglas, steel and air) that demonstrate the wide range of applicability of the method. Table 2.3-1 summarizes these experiments.

-- The average K_{eff} of the benchmarks is 0.9998 which demonstrates that there is no bias associated with the method. The standard deviation of the K_{eff} values is 0.0057 Δk. The 95/95 one sided tolerance limit factor for 27 values is 2.26. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.013 Δk.

The total uncertainty to be added to a criticality calculation is:

$$TU = [(KS)^2_{method} + (KS)^2_{nominal} + (KS)^2_{mech}]^{1/2}$$

where $(KS)_{method}$ is 0.013 as discussed above, $(KS)_{nominal}$ is the statistical uncertainty associated with the particular KENO calculation being used and $(KS)_{mech}$ is the statistical uncertainty associated with mechanical tolerances, such as thicknesses and spacings.

The most important effect on reactivity of the mechanical tolerances is the possible reduction in the water gap between the poison plates. The worst combination of mechanical tolerances are those that result in the maximum reduction in the water gap. For a single can it is found that reactivity does not increase significantly because the increase in reactivity due to the water gap reduction on one side of the can is offset by the decrease in reactivity due to the increased water gap on the opposite side of this can. The analysis, for the effect of mechanical tolerances, however, assumed a worst case of a rack composed of an array of groups of four cans with the minimum water gap between the four cans. The reactivity increase of this configuration is included as a bias term in calculating the K_{eff} of the rack. It is included as a bias term since cans can be welded to a common grid during manufacturing which is the likely cause of the water gap reduction.

An additional reactivity consideration is due to can bowing. The individual can bowing tolerance could also result in a reduction of the water gap between poison plates. Again an array of groups of four assemblies is assumed with the minimum water gap between the four cans. The resulting reactivity increase is included as an uncertainty because can bowing will be random as opposed to the cans welded to a common grid effect. Also, since this common grid effect is already included in the analysis, it is equally likely that can bowing will cause a reactivity decrease as increase from this starting point.

Some mechanical tolerances are not included in the analysis because worst case assumptions are used in the nominal case analysis. An example of this is eccentric assembly position. Calculations were performed which show that the most reactive condition is the assembly centered in the can which is assumed in the nominal case. Another example is the reduced width of the poison plates. No bias is included here since the nominal KENO case models the reduced width explicitly.

The final result of the uncertainty analysis is that the criticality design criteria are met when the calculated effective multiplication factor, plus the total uncertainty (TU) and any biases, is less than 0.95.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurizer Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 5.1.12; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety," NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage;" and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

2.3.2.5 Rack Modification

The spent fuel storage rack is described in Section 2.0. The minimum ^{10}B loading in the poison plates is 0.03 gm $^{10}B/cm^2$.

For normal operation and using the method in the above sections, the K_{eff} for the rack is determined in the following manner:

$$K_{eff} = K_{nominal} + B_{mech} + B_{method} + B_{part} + \quad (ks_{nominal})^2 + (ks_{method})^2 + (ks_{mech})^2 \quad 1/2$$

where:

$K_{nominal}$ = nominal case KENO K_{eff}

B_{mech} = K_{eff} bias to account for the fact that mechanical tolerances can result in water gaps between poison plates less than nominal.

B_{method} = method bias determined from benchmark critical comparisons.

B_{part} = bias to account for poison particle self-shielding.

$ks_{nominal}$ = 95/95 uncertainty in the nominal case KENO K_{eff}.

ks_{method} = 95/95 uncertainty in the method bias.

ks_{mech} = 95/95 uncertainty to account for thickness, spacing and bowing tolerances which are assumed to reduce the water gap between poison plates.

Substituting calculated values, the result is:

K_{eff} = 0.9411

Since K_{eff} is less than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met.

2.3.2.6 Acceptance Criteria For Criticality

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions.

Generally, the acceptance criteria for postulated accident conditions can be $K_{eff} \leq 0.98$ because of the accuracy of the methods used coupled with the low probability of occurrence. For instance, in ANSI N210-1976 the acceptance criteria for the "optimum moderation" condition is $K_{eff} \leq 0.98$. However, for storage pools, which contain dissolved boron, the use of the realistic initial conditions ensures that $K_{eff} << 0.95$ for postulated accidents as discussed in Section 2.3.2.3. Thus, for simplicity, the acceptance criteria for all conditions will be $K_{eff} \leq 0.95$.

2.3.3 Thermal-Hydraulic

The purpose of thermal-hydraulic analysis is to determine the maximum fuel clad temperatures which may occur as a result of using the poison spent fuel racks in the Oconee spent fuel pool.

2.3.3.1 Criteria

The criteria used to determine the acceptability of the design from a thermal-hydraulic viewpoint is summarized as follows:

1. The design must allow adequate cooling by natural circulation and by flow provided by the spent fuel pool cooling system. The coolant should remain subcooled at all points within the pool when the cooling system is operational. When the cooling system is postulated to be inoperable, adequate cooling implies that the temperature of the fuel cladding should be sufficiently low that no structural failures would occur and that no safety concerns would exist.

2. For normal operations, the maximum pool temperature shall not exceed 150°F. For conservatism, the temperatures of the storage racks and the stored fuel are evaluated assuming that the temperature of the

water at the inlet to the storage cells is 150°F during normal operation.

3. The rack design must not allow trapped air or steam. Direct gamma heating of the storage cell walls and the intercell water must be considered.

2.3.3.2 Key Assumptions

1. The nominal water level is 24 feet above the top of the fuel storage racks.

2. The maximum fuel assembly decay heat output is 7.92×10^4 watts.

3. The maximum temperature of the water at the inlet to the storage cells is 150°F when the cooling system is operational.

4. Under postulated accident conditions, when no pool cooling systems are operational, the maximum temperature at the inlet to the cells is assumed to be equal to the saturation temperature at atmospheric pressure or 212°F.

2.3.3.3 Description of Analytical Method and Types of Calculations Performed

A natural circulation calculation is employed to determine the thermal-hydraulic conditions within the spent fuel storage cells. The model used assumes that all downflow occurs in the peripheral gap between the pool walls and the outermost storage cells and all lateral flow occurs in the space between the bottom of the racks and the bottom of the pool. The effect of flow area blockage in the region is conservatively accounted for and a multi-channel formulation is used to determine the variation in axial flow velocities through the various storage cells. The hydraulic resistance of the storage cells and the fuel assemblies is conservatively modeled by applying large uncertainty factors to loss coefficients obtained from various sources. Where necessary, the effect of Reynolds Number on the hydraulic resistance is considered and the variation in momentum and elevation head pressure drops with fluid density is also determined.

The solution is obtained by iteratively solving the conservation equations (mass, momentum and energy) for the natural circulation loops. The flow velocities and fluid temperatures that are obtained are then used to determine the fuel cladding temperatures. An elevation view of a typical model is sketched in Figure 2-5 where the flow paths are indicated by arrows. Note that each cell shown in that sketch actually corresponds to a row of cells that are located at the same distance from the pool walls. This is more clearly shown in a plan view, Figure 2-6.

As shown in that sketch, the lateral flow area underneath the storage cells decreases as the distance from the wall increases. This counteracts the decrease in the total lateral flow that occurs because of flow that branches up and flows into the cells. This is significant because the lateral flow velocity affects both the lateral pressure drop underneath the cells and the turning losses that are experienced as the flow branches up into the cells. These effects are considered in the natural circulation analysis.

The most recently discharged or "hottest" fuel assemblies are assumed to be located in various rows during different calculations in order to ensure that they may be placed anywhere within the pool without violating safety limits. In order to simplify the calculations, each row of the model must be composed of storage cells having a uniform decay heat level. This decay heat level may or may not correspond to a specific batch of fuel, but the model is constructed so that the total heat input is correct. The "hottest" fuel assemblies are all assumed to be placed in a given row of the model in order to ensure that conservatively accurate results are obtained for those assemblies. In fact, the most conservative analysis that can be performed is to assume that all assemblies in the pool (or rows in the model) have the same decay heat rate. This maximizes the total natural circulation flowrate which leads to conservatively large pressure drops in the downcomer and lateral flow regions which reduces the driving pressure drop across the limiting storage locations. This is the approach that has been used to perform the analysis for the Oconee spent fuel storage racks.

Since the natural circulation velocity strongly affects the temperature rise of the water and the heat transfer coefficient within a storage cell, the hydraulic resistance experienced by the flow is a significant parameter in the evaluation. In order to minimize the resistance, the design of the inlet region of the racks has been chosen such as to maximize this flow area. Each storage cell has one or more flow openings as shown in Figure 2-7. The use of these large or multiple flow holes virtually eliminates the possibility that all flow into the inlet of a given cell can be blocked by debris or other foreign material that may get into the pool. In order to determine the impact of a partial blockage on the thermal-hydraulic conditions in the cells, an analysis is also performed for various assumed blockages.

The analyses that have been described only address the flow through the storage cells. As noted in the discussion of criteria, it is also required that the flow and temperatures in the axial gap between adjacent storage cells be evaluated. In order to preclude the possibility of stagnant conditions in these gaps, flow relief areas are provided at the location of the grid support structures as shown in Figure 2-8. This flow area also ensures that air or steam cannot be trapped in the rack structure. The thermal hydraulic conditions in the gap region are evaluated by using a parallel path thermal-hydraulic model of the gap and cell under consideration. This analysis considers the gamma heat generation in the cell enclosure, poison material and cell wrapper in addition to the decay heat input. Using the cell flow velocity and driving pressure differential obtained from the previously described pool analyses, the flow velocity in the gap and the axial temperature distributions of the coolant and structure are determined. The radial temperature distributions through the various components are also considered.

2.3.3.4 Results

<u>Normal Operation</u>

Basis: a) Cooling System Operational

b) 3 days after shutdown-Decay Heat = 75.1 BTU/second/assembly

c) Uniform decay heat loading in pool - No credit for lower actual heat input

d) Peak Rod has 60 percent more heat output than average rod

e) All storage cells filled.

Results of the analysis show that no boiling occurs at any point within the storage racks when the normal cooling system is in operation or whenever pool temperature is maintained within its allowable limits. Water termperatures in the gap between cells are lower than inside the cells, and boiling does not occur in the inter-cell gaps. Although the normal water level is 24 feet above the top of the racks, a level of only 10 feet is required for a saturation temperature of 225°F which is greater than the cell outlet temperature, and no boiling occurs.

Flow Blockage Analysis

Basis: a) 3 days after shutdown

b) Temperature of water at inlet storage racks = 150°F

Results of the analysis show that should up to 75 percent flow blockage occur, there would be no boiling in the water channels between the cells or inside the cells. Because of the large or multiple flow openings that are used in the Westinghouse storage racks, it is very improbable that a complete blockage could occur.

Abnormal Condition

Although it is highly unlikely that a complete loss of cooling capability could occur, the racks are analyzed to this condition.

Basis: a) No pool cooling implies that temperature of water at inlet to spent fuel racks is 212° which corresponds to the saturation temperature at the pool surface.

b) The nominal water level of 24 feet above the top of the racks is maintained.

c) A conservative fuel loading case is assumed. The pool is completely filled with fuel based on a full core discharge at one month following a normal refueling. Previous refuelings of 1/3 core are assumed to have occurred at 1 year intervals.

d) The assemblies that are evaluated are initially put into the pool at 3 days after shutdown.

e) The peak rods are assumed to have 60 percent greater heat output than average rods.

f) All storage cells are filled and all downflow occurs in the peripheral gap.

Results of this analysis show that due to the effects of natural circulation, the fuel cladding temperatures are sufficiently low to preclude structural failures. No boiling in the water channels between the fuel assemblies and within the storage cells occurs.

Since the saturation temperature is approximately 239°F and the maximum cell outlet temperature at 3 days after shutdown is about 234°F boiling does not occur in the water channels between fuel assemblies. As decay heat decreases, the cell outlet temperatures also continue to decrease.

2.3.4 Neutron Absorbing Material

The neutron material, Boraflex, used in the Oconee spent fuel rack construction is manufactured by Brand Industrial Services, Inc., and fabricated to safety related nuclear criteria of 10CFR50, Appendix B. Boraflex is a silicone based polymer containing fine particles of boron carbide in a homogeneous, stable matrix. Boraflex contains a minimum ^{10}B areal density of 0.03 gm/cm^2.

Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material.(7) Tests were performed at the University of Michigan exposing Boraflex to 1.03 x 10" rads gamma radiation with a substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities before and after being subjected to an environment of borated water and 1.03 x 10" rads gamma radiation.(8)

Long term borated water soak tests at high temperatures were also conducted.(9) It was shown that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. Boraflex maintains its functional performance characteristics and shows no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

During irradiation, a certain amount of gas may be generated. A conservative evaluation of the effect of gas generation on the spent fuel pool building atmosphere indicates that the maximum gas generation would be less than 0.01 percent of the total room volume. Additionally, the majority of gas generation is nitrogen, oxygen and CO_2.

The actual tests verify that Boraflex maintains long-term material stability and mechanical integrity, and can be safely utilized as a poison material for neutron absorption in spent fuel storage racks.

2.3.5 Spent Fuel Rack Surveillance Program

The following section provides a general description of the surveillance program Duke Power Company plans to implement with respect to the spent fuel racks being proposed for Oconee Unit 3 spent fuel pool. The purpose of this surveillance program is to assure the mechanical integrity and neutron absorption capability of the Boraflex neutron poison material used in the racks is maintained. The program described below is based on current performance information on the Boraflex material. However, in the coming years, the nuclear industry will gain more information on the performance of Boraflex through both experimentation and operating experience. Duke will evaluate this information as it becomes available and will modify the surveillance program as determined warranted and justified.

Proper documentation will be obtained from the manufacturers of the Boraflex and the racks to assure the quality of the neutron poison material and its proper loading in the racks. Duke will perform a visual inspection of the racks upon receipt to verify that the Boraflex is loaded in each of the specified locations in the rack.

A representative sampling of Boraflex specimens will be selected from the lots of material used in the fabrication of the racks. Although the exact number of specimens which will be used is still being evaluated by Duke, it is expected that a minimum of 25 specimens will be used. Each specimen will be placed in a stainless steel holder and immersed in the spent fuel pool. The specimens will be located within the spent fuel pool such that they will receive exposure to a representative gamma flux.

Irradiation tests have been previously performed to test the stability of Boraflex in boric acid solution and under irradiation. The results of these tests are documented in Bisco test reports.[7,8,9] From these tests, there is no evidence indicating any deterioration of the Boraflex material through a cumulative irradiation in an excess of 1×10^{11} rads gamma effecting the suitability of Boraflex as a neutron shielding material. Duke has calculated that the specimens would require at least 10 years in the pool environment to approach this level of cumulative exposure.

Duke plans to perform an initial surveillance of the specimens after approximately five years of exposure in the pool environment. During this surveillance several specimens will be removed from the pool and checked for mechanical integrity as well as absorption capability. This examination is expected to include visual inspection as well as other tests determined necessary to verify the material stability. This initial surveillance will be used to verify that the performance of the Boraflex is consistent with the Bisco test results. Based on the results of this initial surveillance, and results from the Unit 1 and 2 spent fuel rack surveillance program, Duke will determine the scheduling and extent of additional surveillances so as to assure acceptable material performance throughout the life of the plant.

TABLE 2.3-1

BENCHMARK CRITICAL EXPERIMENTS[4,5,6]

	General Description	Enrichment w/o U235	Reflector	Separating Material	Characterizing Separation (cm)	K_{eff}
1.	UO_2 rod lattice	2.35	water	water	11.92	1.004 ± .004
2.	UO_2 rod lattice	2.35	water	water	8.39	0.993 ± .004
3.	UO_2 rod lattice	2.35	water	water	6.39	1.005 ± .004
4.	UO_2 rod lattice	2.35	water	water	4.46	0.994 ± .004
5.	UO_2 rod lattice	2.35	water	stainless steel	10.44	1.005 ± .004
6.	UO_2 rod lattice	2.35	water	stainless steel	11.47	0.992 ± .004
7.	UO_2 rod lattice	2.35	water	stainless steel	7.76	0.992 ± .004
8.	UO_2 rod lattice	2.35	water	stainless steel	7.42	1.004 ± .004
9.	UO_2 rod lattice	2.35	water	boral	6.34	1.005 ± .004
10.	UO_2 rod lattice	2.35	water	boral	9.03	0.992 ± .004
11.	UO_2 rod lattice	2.35	water	boral	5.05	1.001 ± .004
12.	UO_2 rod lattice	4.29	water	water	10.64	0.999 ± .005
13.	UO_2 rod lattice	4.29	water	stainless steel	9.76	0.999 ± .005
14.	UO_2 rod lattice	4.29	water	stainless steel	8.08	0.998 ± .006
15.	UO_2 rod lattice	4.29	water	boral	6.72	0.998 ± .005
16.	U metal cylinders	93.2	bare	air	15.43	0.988 ± .003
17.	U metal cylinders	93.2	paraffin	air	23.84	1.006 ± .005
18.	U metal cylinders	93.2	bare	air	19.97	1.005 ± .003
19.	U metal cylinders	93.2	paraffin	air	36.47	1.001 ± .004
20.	U metal cylinders	93.2	bare	air	13.74	1.005 ± .003
21.	U metal cylinders	93.2	paraffin	air	23.48	1.005 ± .004
22.	U metal cylinders	93.2	bare	plexiglas	15.74	1.010 ± .003
23.	U metal cylinders	93.2	paraffin	plexiglas	24.43	1.006 ± .004
24.	U metal cylinders	93.2	bare	plexiglas	21.74	0.999 ± .003
25.	U metal cylinders	93.2	paraffin	plexiglas	27.94	0.994 ± .005
26.	U metal cylinders	93.2	bare	steel	14.74	1.000 ± .003
27.	U metal cylinders	93.2	bare	plexiglas steel	16.67	0.006 ± .003

FIGURE 2-1

FUEL STORAGE RACK MODULE

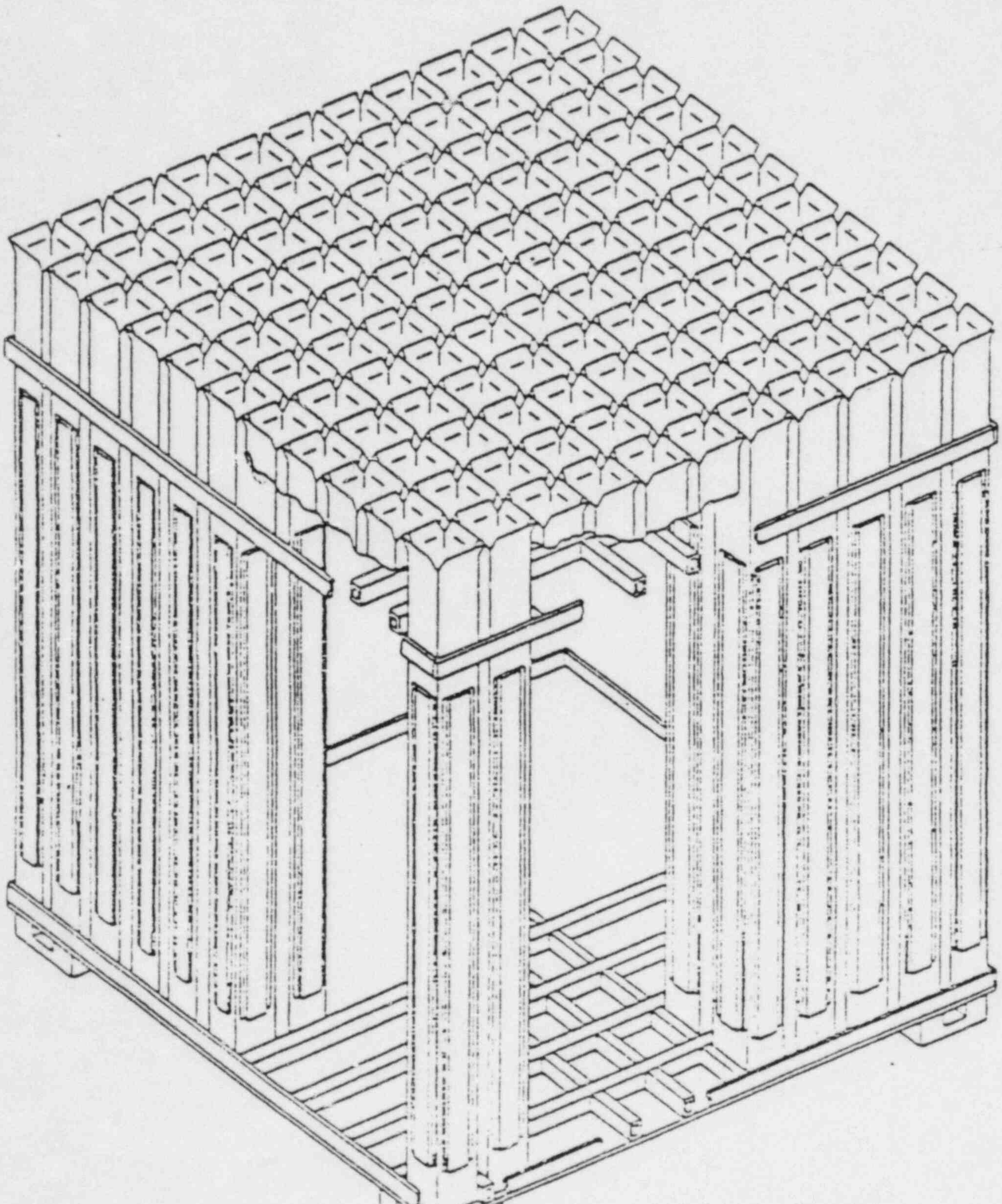


FIGURE 2-1

POOL OUTLINE AND RACK ARRANGEMENT

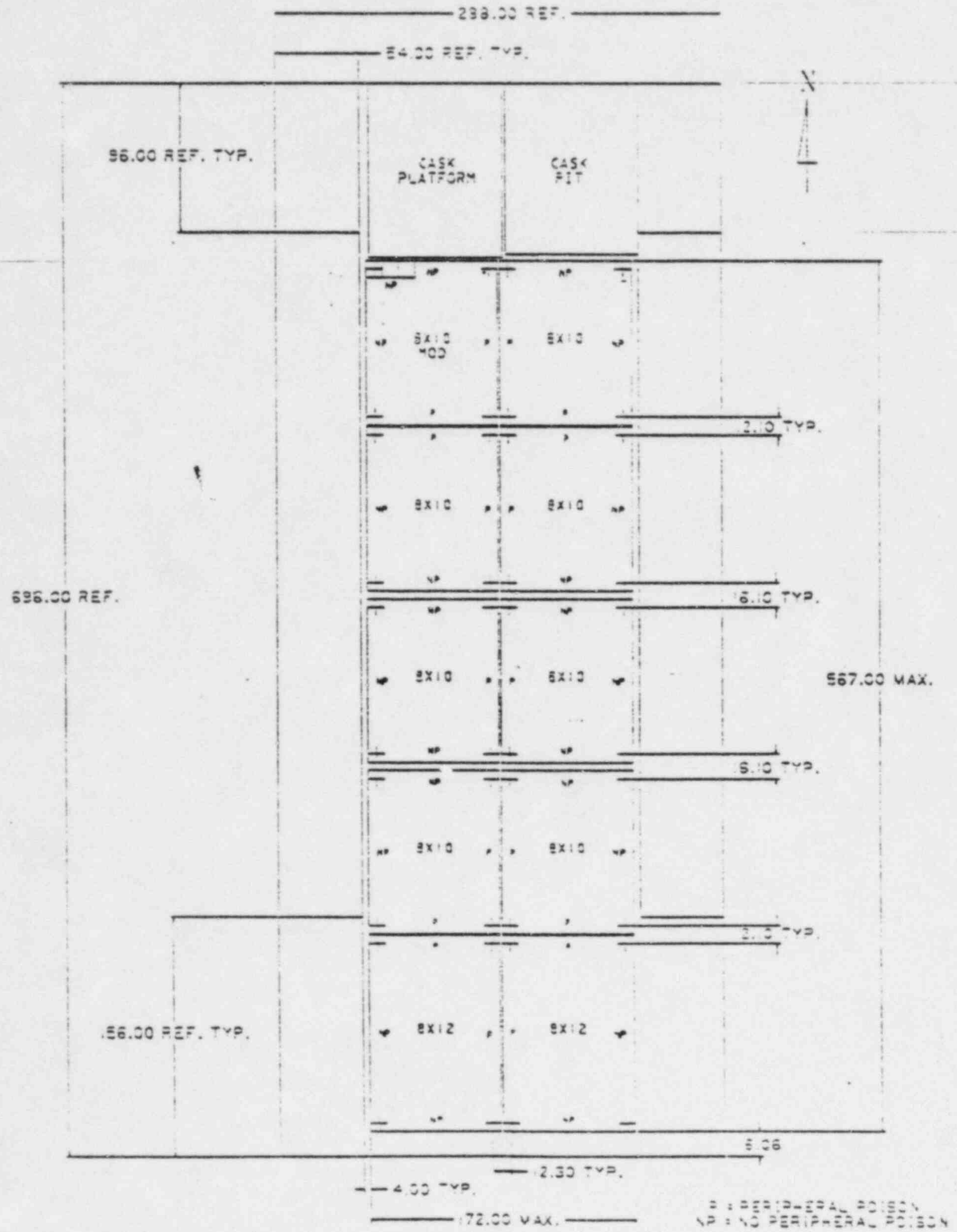
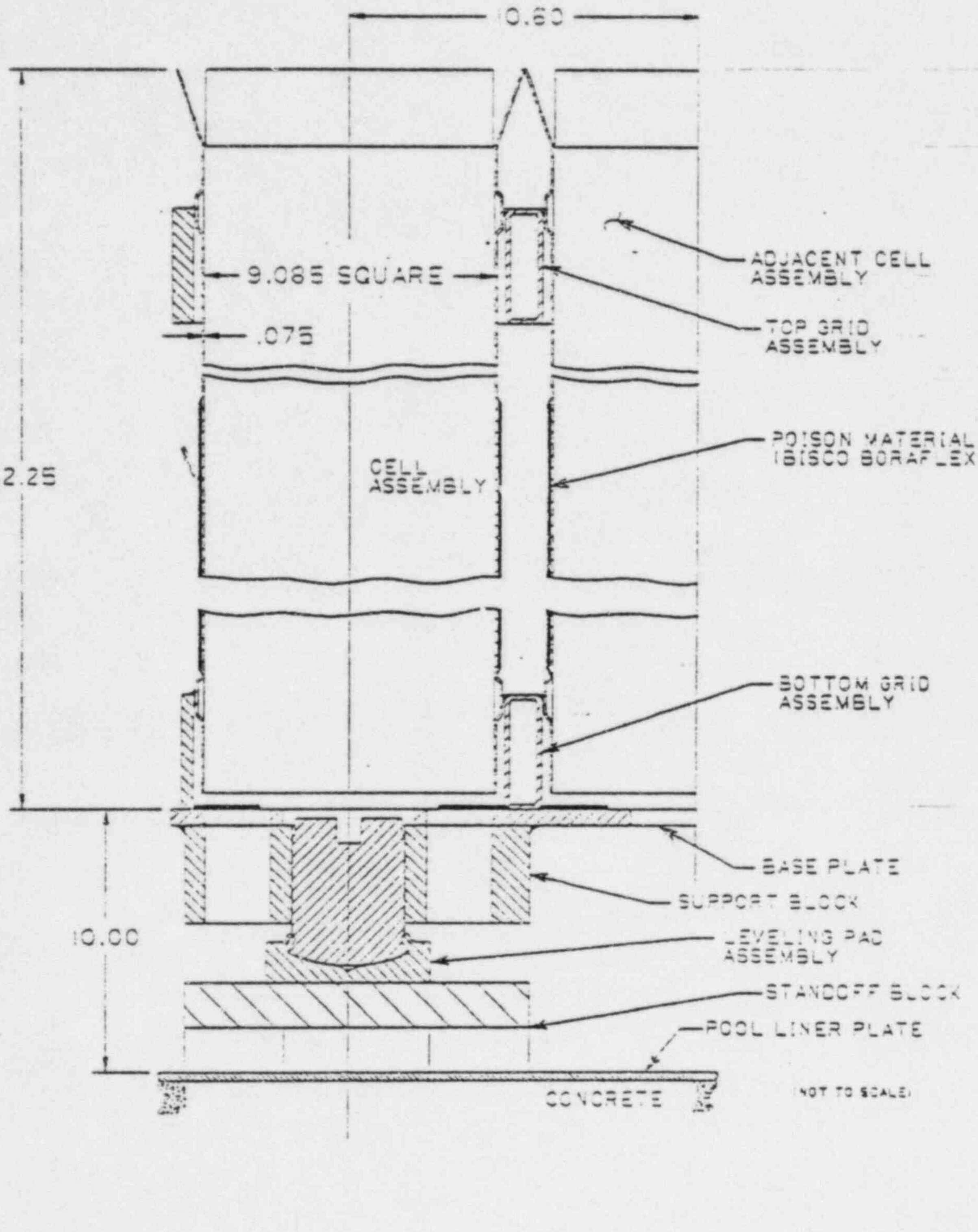


FIGURE 2-3

FUEL RACK ASSEMBLY



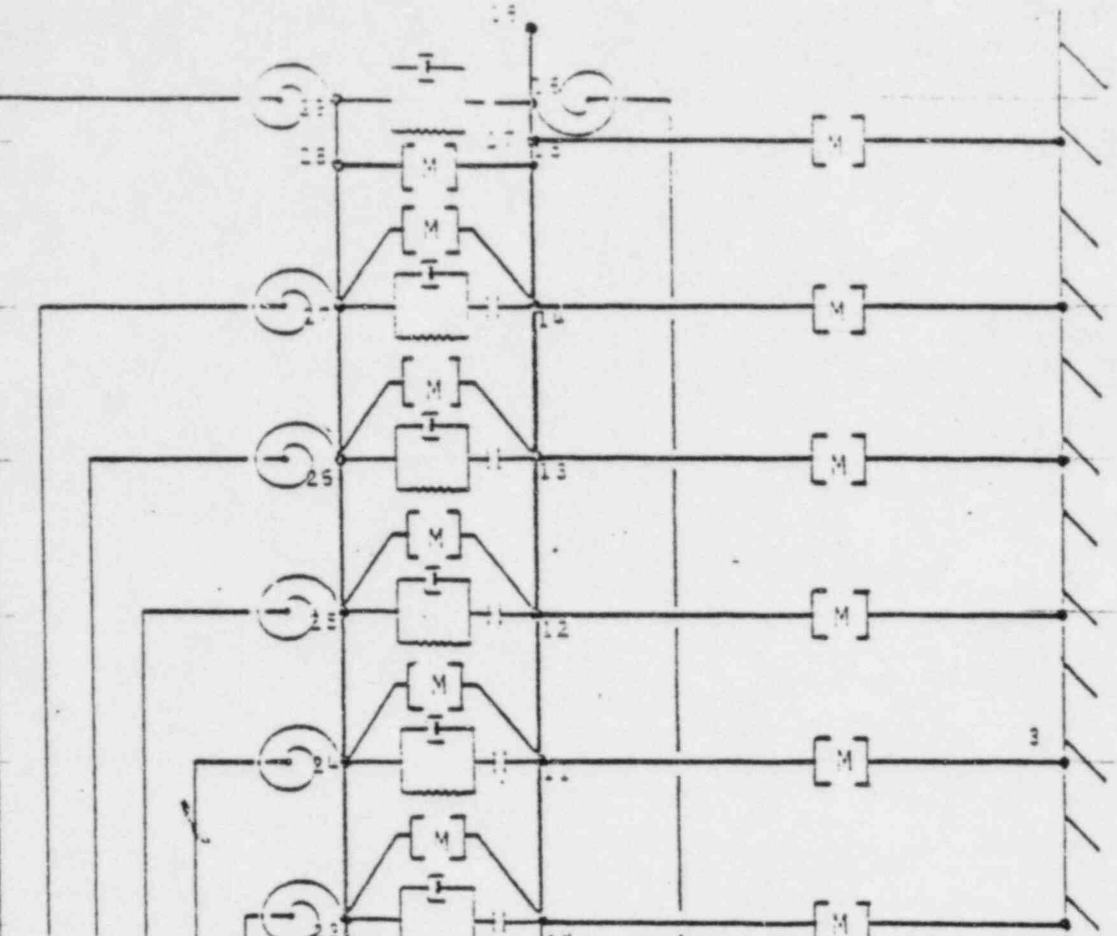


FIGURE 2-4(A) NONLINEAR SEISMIC MODEL

FIGURE 2-3

DETAIL SEISMIC MODEL

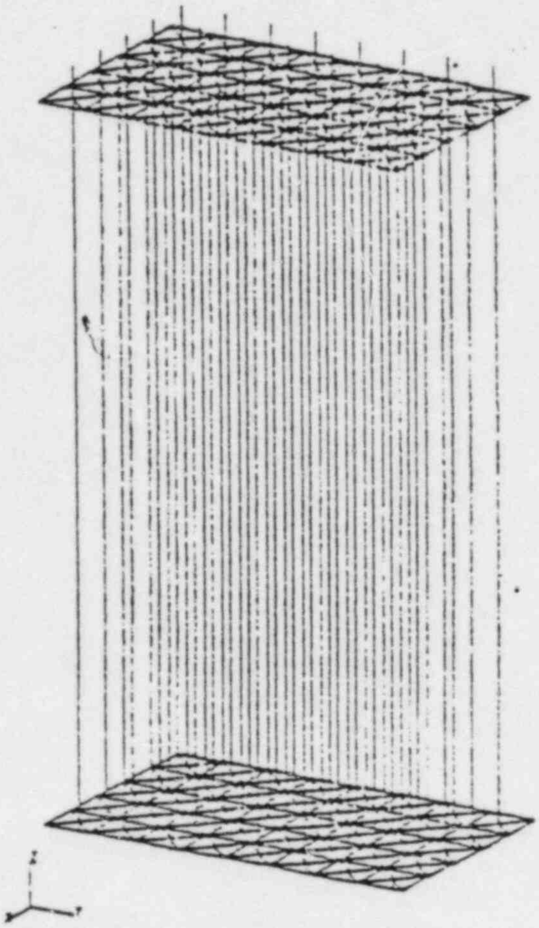


FIGURE 2-6

SPENT FUEL POOL NATURAL CIRCULATION MODEL
(ELEVATION VIEW)

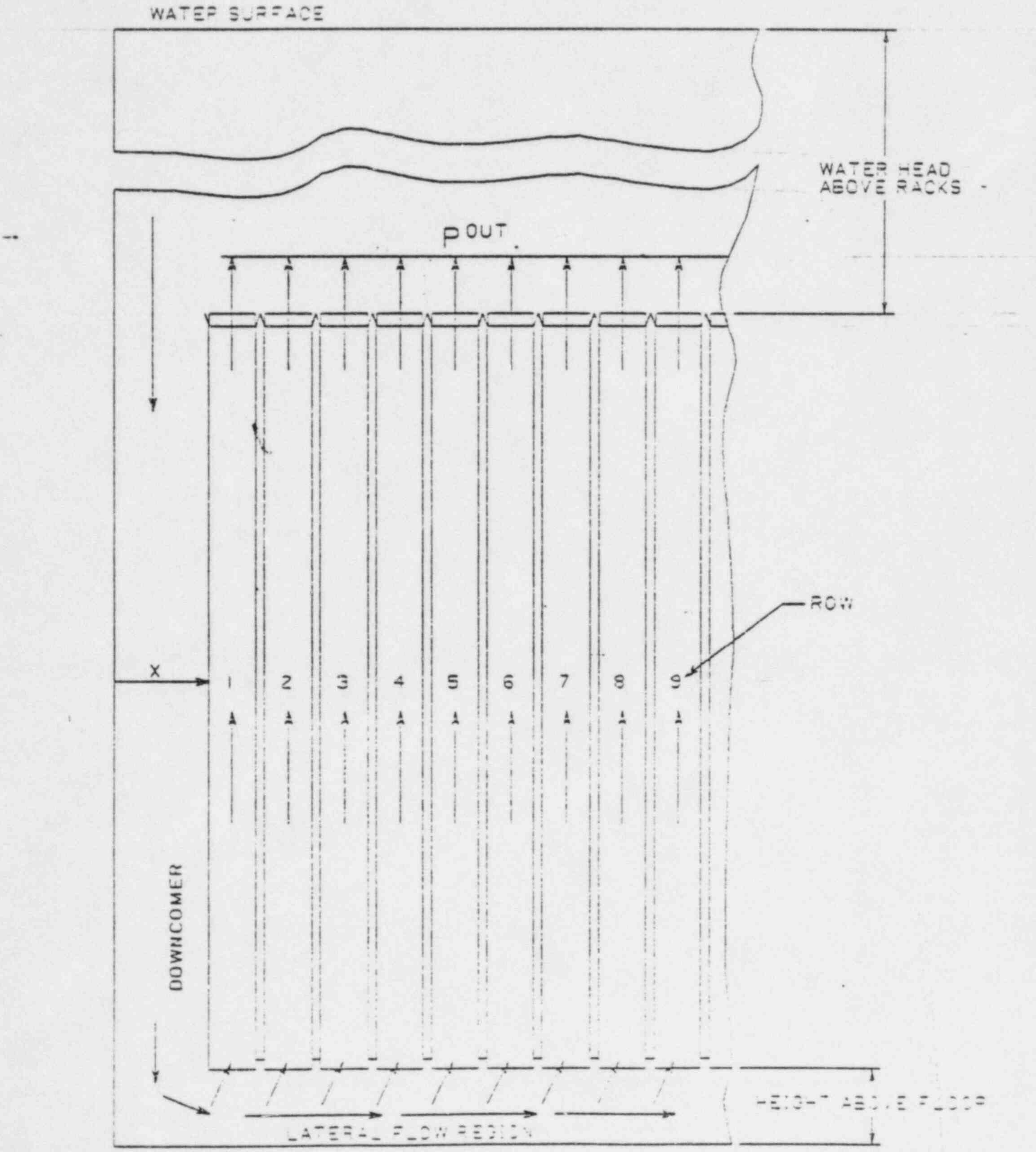


FIGURE 2-6

SPENT FUEL POOL NATURAL CIRCULATION MODEL
PLAN VIEW

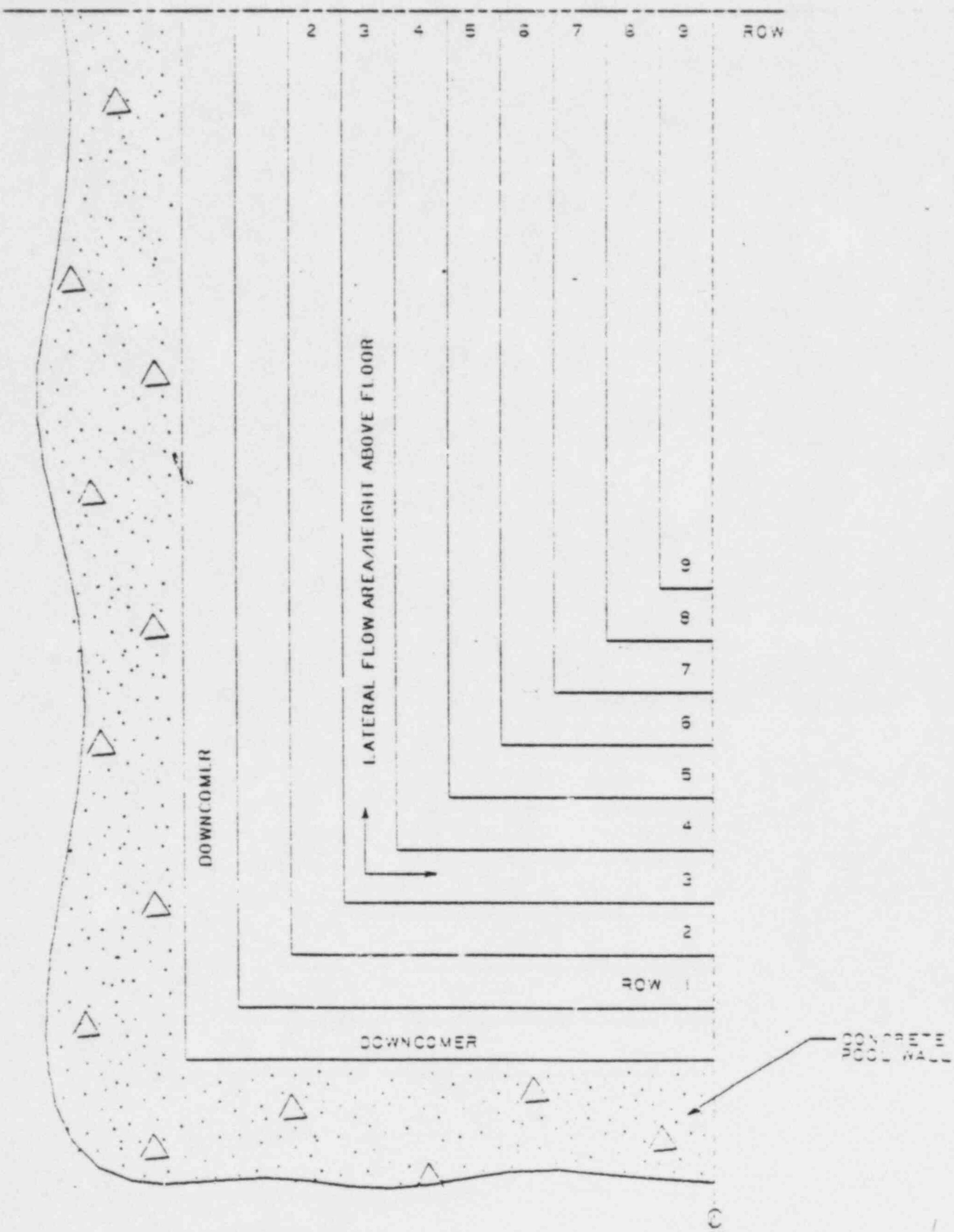


FIGURE 2-7

SPENT FUEL RACK INLET FLOW AREAS

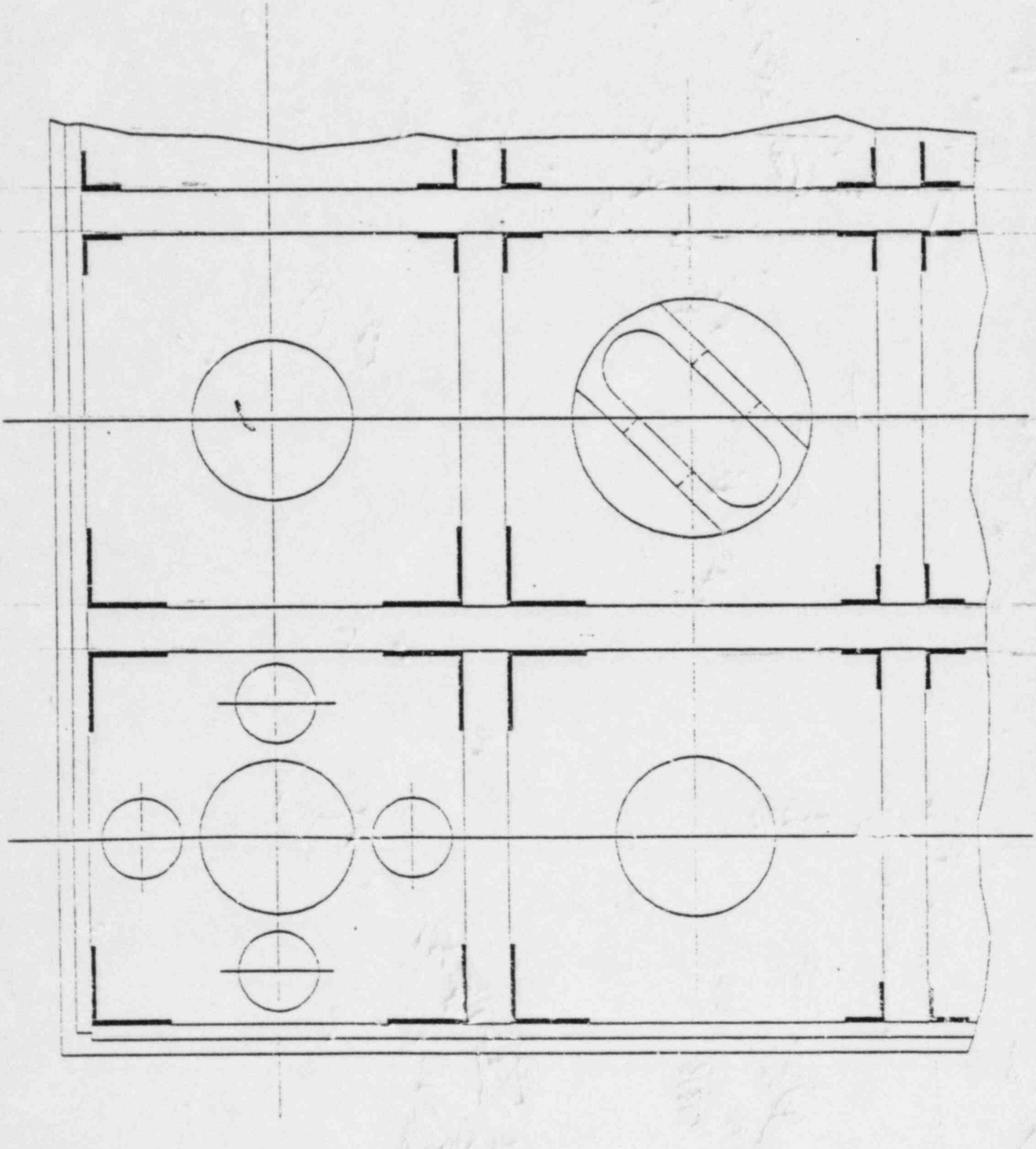
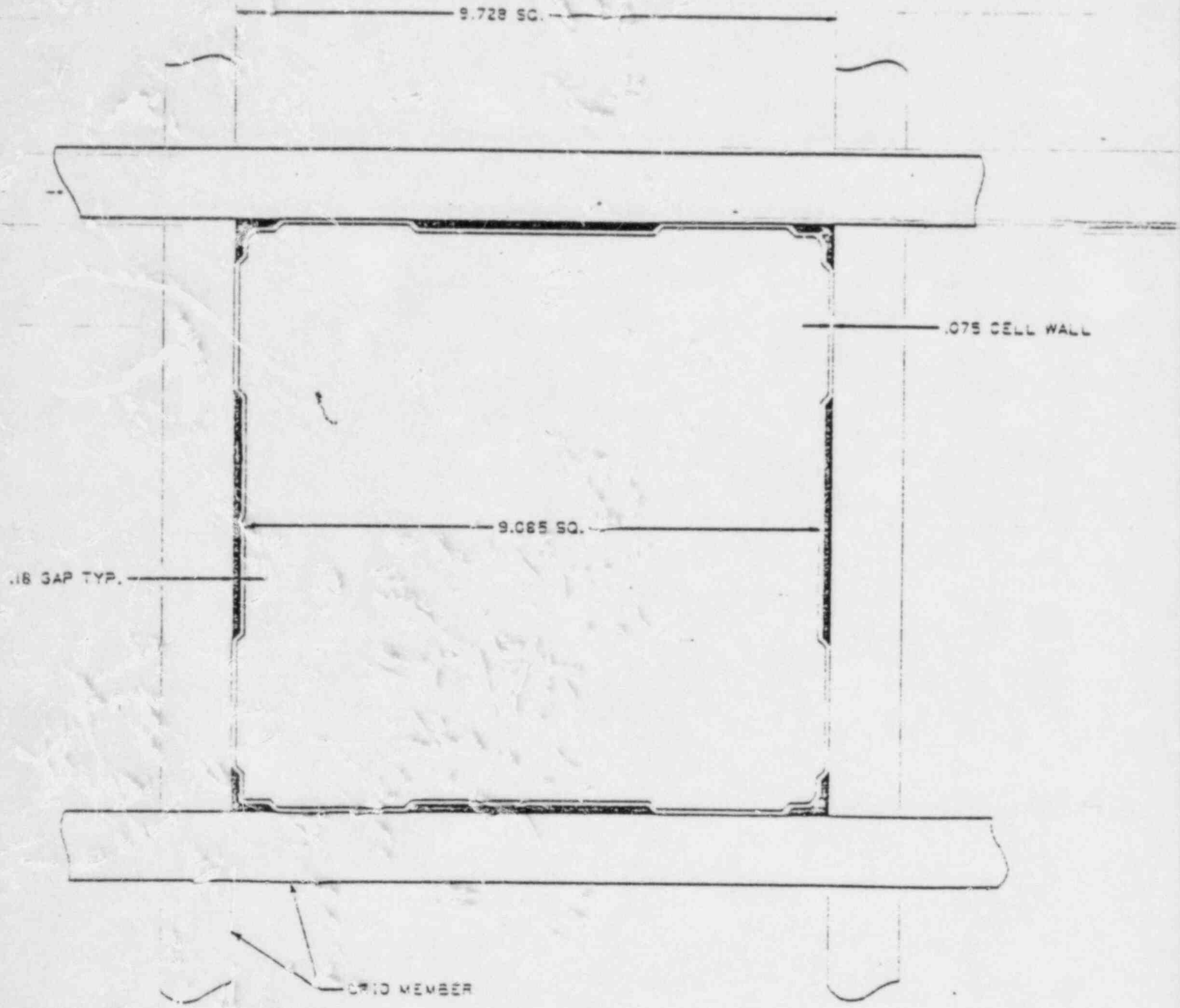


FIGURE 2-6

INTERCELL FLOW AREA REGION



GAP FLOW AREA AT SUPPORT = 2.34 IN.^{2}

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8. J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1, (July 1979).

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3.0 SPENT FUEL INTERFACE

3.1 STRUCTURAL

The spent fuel pool and its cooling system are described in the Oconee Nuclear Station Final Safety Analysis Report Section 9.1.3. The general arrangement of the Unit 3 pool and the associated fuel handling equipment is not changed as a result of this modification. However, an additional cooling train will be provided before the quantity of stored fuel assemblies exceeds the previously licensed capacity (474 assemblies).

The spent fuel pool is constructed of reinforced concrete lined with stainless steel plate. The fuel pool concrete, reinforcing steel, linear plate and welds connecting the liner plate to the fuel pool floor concrete embedments are analyzed based on consideration of the new racks and additional fuel. Design criteria including loading combinations and allowable stresses are in compliance with Oconee FSAR Section 3.8.4 for Class I structures. The determination of Ta (abnormal thermal load condition to be used in combination with E') is based on the failure of one pump or cooler during normal operating conditions.

The rack/spent fuel pool interface is described in Section 2.2.

3.2 THERMAL

3.2.1 Design Bases

As specified in the Oconee FSAR Section 9.1.3, the spent fuel pool and pool cooling system are designed to maintain the pool water temperature at 150°F or less for normal refueling operations and full core discharge situation will all pumps and coolers operating, and at 205°F or less with postulated loss of one pump or cooler. With the addition of a third pump and cooler train, these criteria established in the Oconee FSAR will be met. Under normal refueling conditions the fuel is discharged over a four day period after at least three days cooling inside the reactor vessel. The full core discharge is expected to take four days also with three days cooling in the reactor vessel prior to moving any fuel. The heat released from the fuel stored in the pool is determined in accordance with both the Standard Review Plan (SRP-9.1.3) "Spent Fuel Pool Cooling and Clean-up System" and Oconee FSAR Section 9.1.3.2.3.1. Table 3.2.1 shows both results of the heat loading in the pool, which are consistant. In the event mixed oxide fuel becomes available the heat load in the pool will be slightly higher. The increase is apparent only in fuel which has decayed for a relatively long period of time and contributes little additional heat load to the pool.

3.2.2 System Description

The Spent Fuel Cooling System is described in the Oconee FSAR Section 9.1.3.2. This system will be augmented by the addition of a third spent fuel cooler and pump which will take suction from the existing spent fuel pool coolant piping. Heat exchanger cooling water will be drawn from the recirculating water system. The added cooler and pump are described in Table 3.2.2.

3.2.3 Design Evaluation

During normal operation the Spent Fuel Cooling System serves two main functions. The first is to maintain the pool water at temperatures below 150°F. The second function is to provide purification of the spent fuel pool coolant for clarity during fuel handling operations. When installation is complete, the three pump and cooler trains will be arranged in parallel. The purification function is performed as described in the Oconee FSAR Section 9.1.3.2.3.1.

The heat loads shown in Table 3.2.1 represent the heat loads expected in the spent fuel pool as calculated in accordance with 1) the Oconee FSAR Section 9.1.3.2.3.1 criteria and 2) the NRC Standard Review Plan (SRP-9.1.3). The postulated inventory, as assumed in the FSAR, is one full core discharge with the remainder of the storage locations occupied by batches previously discharged at one-year intervals. The maximum case, as specified in SRP-9.1.3, assumed a normal refueling discharge followed by a full core discharge after a short period of operation. For the normal case, it is assumed that Unit 3 has been refueled and the pool is filled with two previous discharges.

In accordance with FSAR Section 9.1.3.2.3.1, the spent fuel pool temperature under maximum heat load conditions will be maintained below 150°F by operation of all three pumps and coolers. Upon failure of one pump or cooler sufficient cooling capacity will remain to maintain bulk pool temperature below 205°F. In addition, analysis was performed in accordance with the criteria established by the Standard Review Plan. It was found that with the loss of one pump or cooler for the normal heat load case, sufficient cooling capacity remains to maintain the spent fuel pool temperature below the specified criterion of 140°F. It was also shown that for the maximum heat load case, with all three trains operating, the bulk pool temperature will be below 150°F and thus will not reach the criterion of boiling. An analysis of pool response to loss of all forced cooling is presented in Section 6.3 of this document.

Table 3.2-1

Heat Loads for the Unit 3 Spent Fuel Pool Rerack

	FSAR Criteria	Criteria
Normal Heat Load	—	12.6×10^6 Btu/hr
Maximum Heat Load	30.4×10^6 Btu/hr	30.8×10^6 Btu/hr

Table 3.2-2

Components to Be Added to Oconee Unit 3 Spent Fuel Pool Cooling System

Spent Fuel Cooler

Type	Plate
Material	Stainless Steel
Normal Capacity, BTU/hr/cooler	7.75×10^6
Code	ASME Section III-3

Spent Fuel Pump

Type	Horizontal Centrifugal
Material	Stainless Steel
Flow, gpm	1000
Code	ASME Section III-3

3.3 WATER QUALITY

Operating experience has shown that concentrations of radionuclides are greatest during periods of fuel movement in the pool (i.e., refueling) and are not directly related to the number of assemblies stored in the pool. Therefore, the increased load on the Spent Fuel Pool Purification System will be small and the existing system will adequately maintain water chemistry, clarity, and activity within acceptable levels.

4.0 RACK INSTALLATION

The installation plan is based on the following objectives:

a) Maintaining installation exposure levels as low as reasonably achievable (ALARA).

b) Removal of all fuel assemblies from the Oconee 3 pool prior to commencement of reracking operation.

c) Achieving acceptable tolerances on module verticality, levelness, and positioning.

4.1 QUALITY ASSURANCE

-- The quality assurance aspects of the removal of the existing racks and the installation of the new racks will be carried out in such a manner as to meet the applicable requirements of the Duke Power Company Quality Assurance Program as described in Topical Report DUKE-1A.

4.2 REMOVAL OF EXISTING RACKS

The existing storage racks are Combustion Engineering, Inc., Supplied High Capacity (Hi-Cap) Fuel Assembly Storage Racks. The Hi-Cap Fuel Assembly Storage Racks are constructed of type 304 stainless steel. The configuration of these racks is shown in Figure 4-1. All ten modules are interconnected and rest on the pool floor.

The removal of the existing modules will be accomplished as follows:

a) Removal of all spent fuel assemblies from the pool.

b) Installation of a temporary construction (T-C) crane.

c) The first four sets of interconnected modules will be removed by first lifting them to the rack support frame with the T-C crane. Underwater divers will then be used to perform cutting operations to separate interconnected modules. It is intended that all underwater cutting operations will be performed while using an underwater vacuuming system with shielded filters.

d) Each individual module will then be removed by lifting them to the cask platform with the T-C crane. Then, the modules will be rerigged to the 100 ton cask handling crane and moved to the fuel receiving area for packaging. Each module will be hosed down and allowed to dry before it is removed from the pool area.

e) The two interconnected modules located the furthest from the cask storage pit (module location 5, 6) will initially be moved North by use of two lift bags. The interconnected modules are then rerigged to the T-C crane for placement on the rack support frame.

f) Steps c and d are repeated to complete removal of all the modules.

g) Removal of all bearing pads.

Final disposal of the existing racks is discussed in Section 5.

4.3 INSTALLATION OF NEW RACKS

Th final configuration of the 10 new modules, supplied by Westinghouse Electric Corporation, is shown in Figure 4-2. The installation of the new modules will be accomplished as follows:

a) All new modules will be brought into the fuel receiving area preassembled by truck and lifted to the cask platform by the 100 ton cask handling crane.

b) The modules will then be rerigged to the T-C crane. The two southern most modules will be installed first (module locations 5 and 6), by placing them as far south as possible with the T-C crane.

c) The modules are then rerigged to a lift bag for final placement.

d) If these modules can not be properly set using the lift bag, then the spent fuel handling bridge will be removed and final placement accomplished with the T-C crane.

e) All other modules will be moved underwater to their final position with the T-C crane.

The shim plates which are welded to the bottom liner of pool will remain in place. The standoff plates will be positioned on the pool floor in designated locations prior to installing the racks. The standoff plates will be of sufficient height such that the new modules will be positioned above the existing shim plates.

New module verticality and levelness tolerances will be achieved by the use of screw-adjustable supports. Module to module positioning will be verified by measurement. Each module will be checked to insure that verticality, levelness and position are within design tolerances.

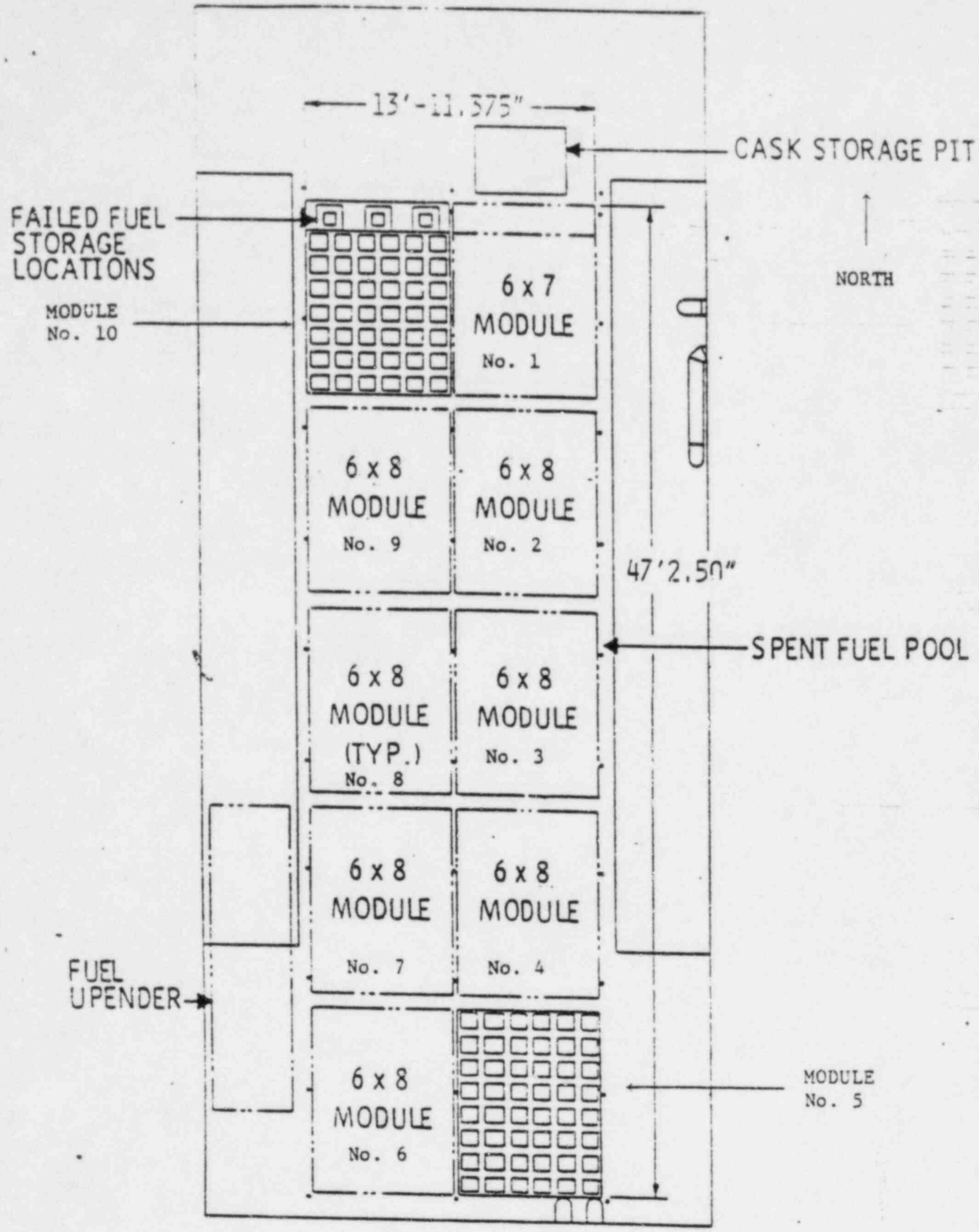
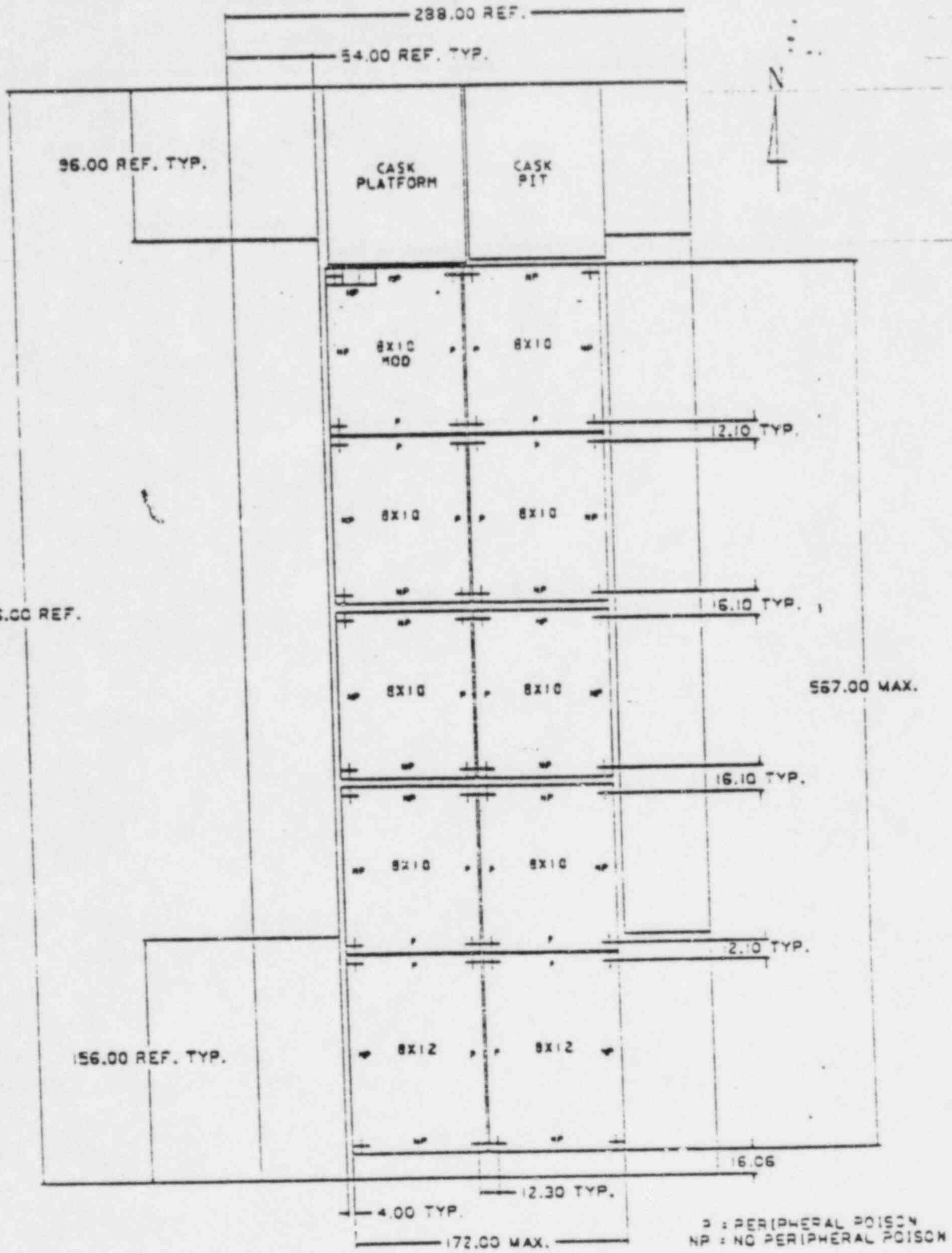


FIGURE 4.1

SPENT FUEL STORAGE AREA FUEL MODULES ARRANGEMENT

FIGURE 4-2

NEW SPENT FUEL RACK ARRANGEMENT



5.0 RADIATION PROTECTION

5.1 GENERAL DESCRIPTION OF RADIOLOGICAL ASPECTS OF PROJECT

The radiation protection aspects of the spent fuel pool modification are the responsibility of the Station Health Physicist, who is assisted by his staff, with the support of the System Health Physicist and his staff. Gamma radiation levels in the pool area are constantly monitored by the station Area Radiation Monitoring System, which has a high level alarm feature. Additionally, periodic radiation and contamination surveys are conducted in work areas as necessary. Where there is a potential for significant airborne radionuclide concentrations, continuous air samplers are used in addition to periodic grab sampling. Personnel working in radiologically controlled areas shall wear protective clothing and respiratory protective equipment, depending on work conditions, as required by the applicable Radiation Work Permit (RWP). Personnel monitoring equipment is assigned to and worn by all personnel in the work area. At a minimum, this equipment consists of a thermoluminescent dosimeter (TLD) and self-reading pocket dosimeter. Additional personnel monitoring equipment, such as extremity badges, shall be worn by divers working in the pool.

Contamination control measures are used to protect persons from internal exposures to radioactive material and to prevent the spread of contamination. Radiation Control Zones (RCZ's) are established around the work area. Work, personnel traffic, and the movement of material and equipment in and out of the area are controlled so as to minimize contamination problems. Material and equipment removed from the SFP will be rinsed, decontaminated further if necessary, and wrapped and/or tagged as necessary. Divers exiting the pool water will also be rinsed off to minimize personnel and area contamination problems. The station radiation protection staff closely monitors and controls all aspects of the work to ensure that personnel exposures, both internal and external, are maintained as low as reasonably achievable (ALARA).

5.1.1 Underwater Radiation Survey

In addition to periodic measurement of dose rates around and above the pool, underwater surveys shall be conducted to determine the dose rates in areas where divers must work or pass through.

A low and high range underwater radiation monitoring instrument will be used, when applicable, to perform dose rate measurement underwater.

5.1.2 Pool Decontamination and Clean-Up

The Spent Fuel Pool Cooling System provides purification and clarification of pool water by recirculating it through a demineralizer and filters. This system operates in this mode to minimize radiation exposure to personnel from the amount of dissolved and suspended radionuclides in the pool water. The water shall be sampled weekly to monitor the concentration of the radionuclides in the pool. In addition, a portable filtered water vacuum system will be used, as necessary, to clean loosely deposited contaminants from the pool floor, walls, and fuel rack surfaces around diver working areas to minimize radiation exposures. A floating skimmer will be available if needed to minimize exposure due to floating crud.

5.1.3 Diving Operations

Prior to all diving operations, the spent fuel assemblies stored in the pool will be removed so as to yield the lowest practicable dose rates to divers and expedite rack replacement. Designated underwater travel paths will be established for divers, as necessary, to ensure that exposures received going to and from the work areas are maintained ALARA. Health Physics personnel will be in the immediate area at all times when divers are in the water. Their duties will be to provide health physics support to minimize personnel exposure and to enforce good radiological work practices and adherence to RWP requirements. They, along with the diver's supervisor, who will be in direct communication with the divers, will continually observe the divers while they are in the pool.

Divers will wear protective clothing items inside their rubber diving suits to protect them from contamination when they remove their diving suits and exit the SFP area. TLD's will be worn inside the diving suits on the head and chest, legs just above the knees, back, and extremities. Self-reading pocket dosimeters will be sealed in plastic bags and also worn inside the diving suit. The self-reading pocket dosimeters will be read and recorded after each dive. A daily tabulation of each individual's cumulative whole body and extremity doses will be prepared on each diver and will be reviewed by the diving supervisor and the cognizant Health Physics Supervisor. This information will be used in part (1) to maintain doses ALARA within the limits and (2) to efficiently allocate exposure among the divers working in the pools.

5.1.4 Decontamination of Removed Rack Sections

When the racks are removed from the spent fuel pool, they will be rinsed with a spray using demineralized water or spent fuel pool water. Personnel involved in this operation or others in the immediate area will wear appropriate protective clothing and respiratory protective equipment, if needed. The rack sections will be allowed to drip dry prior to movement to the new fuel receiving area to be packaged for storage and ultimate disposal. This rinsing operation is expected to remove significant quantities of loose contamination from the racks while causing a relatively low exposure to decontamination personnel. This procedure minimizes subsequent personnel exposures due to handling and packaging of the rack sections for disposal.

5.1.5 Anticipated Exposures During Re-Racking

Table 5.1-1 is a summary of expected exposures for each phase of the reracking operation and for each group of workers. These estimates are based upon a task by task comparison of the man hours and dose rates observed in the 1979 Units 1 and 2 re-racking with the man hours and dose rates anticipated for this re-racking. Adjustments were made based on the proposed installation plan and include 926 total fuel transfers, no fuel present in the pool during reracking, offsite decontamination or disposal.

5.2 DISPOSITION OF OLD RACKS

Burial, decontamination, and long term storage on-site of the racks until reuse or plant decommissioning have been evaluated for the disposal of the ten contaminated racks. The racks will be decontaminated if possible. Depending on the effectiveness of decontamination, the racks will eventually either be sold as scrap or buried at a low-level burial site. If decontamination is not feasible, the racks will be sent off-site for burial.

5.3 SUMMARY OF OCCUPATIONAL DOSE CONSIDERATIONS OF ADDITIONAL SPENT FUEL STORAGE

The occupational exposure for the reracking operation is estimated to be about 22 person-rem. This estimate represents <2% of the average annual station dose. All work will be performed in accordance with a radiation preplan to identify all protection requirements and in a manner consistent with the "as low as reasonably achievable" (ALARA) occupational exposure principle. Health Physics personnel will be available to assure that ALARA considerations prevail.

The estimated increment in occupational dose resulting form the proposed increase in stored fuel assemblies based on present and projected operations in the SFP area is estimated to be less than 1% of the annual station dose. Due to the depth of water shielding the fuel, the additional spent fuel assemblies will contribute a negligible amount to dose rates in the pool area while recirculation of spent fuel pool cooling water through demineralizers and filters will reduce the dissolved and suspended radionuclides present.

TABLE 5.1-1

ESTIMATED ALARA DOSES DURING RE-RACKING
(All doses are in person-rem)

	Rmv. Fuel Bridge	Install + Rmv. Temp. Crane	Fuel Transfer	Vacuum Pool for Rerack	Rack Removal and Replacement	Rack Cutting	Rack Disposal	Total
Operations	0.240	---	2.95	0.125	0.300	---	0.080	3.695
Maintenance	0.480	0.865	0.02	0.055	2.915	2.4	0.875	7.610
Health Physics	0.180	0.070	0.190	0.325	0.955	0.136	0.270	2.126
Engineering	--	0.010	0.032	0.00	0.430	--	--	0.472
Divers	--	--	0.00	0.120	6.180	--	--	6.300
Miscellaneous	0.480	--	0.073	0.048	0.422	0.036	0.072	1.131
TOTAL	1.380	0.945	3.265	0.673	11.202	2.572	1.297	21.334

6.0 SAFETY ANALYSIS

The following analyses are related to postulated accidents associated with operations in and around the spent fuel pool.

6.1 CONSTRUCTION ACCIDENT

There will be no fuel assemblies in the fuel pool during rack installation. Therefore, any construction accident would have no radiological consequence.

6.2 CASK/HEAVY-LOAD ACCIDENT

In order to calculate the consequences of a cask drop accident, it is necessary to determine the maximum number of fuel assemblies which could be contacted. The worst case is considered to be a hoist cable failure when the cask is positioned over the fuel pool wall and the cask has an eccentric drop into the wall. In this case, yoke and load block could be deflected onto the spent fuel.

There are 128 cans under the projected cask, yoke, and block impact area. These cans buckle and deflect into adjacent cans until the total energy of the falling cask is absorbed. In total, 486 cans can potentially suffer a loss of integrity during a cask drop accident.

The radiological consequences of the cask drop accident will be mitigated by limiting the age of fuel stored in the first 31 rows. No cask movement will be allowed if fuel in these locations has decayed less than 70 days. The worst radiological consequences experienced would result from 100% of the activity contained in the fission gases trapped in gaps in the fuel stored in the locations being released into the pool water. The exclusion area boundary dose, taking no credit for ventilation system filtration, would be 0.1 rem whole body and 55 rem to the thyroid. These doses are well below 10 CFR Part 100 limits.

6.3 LOSS OF FORCED COOLING

The large volume of water in the spent fuel pool takes several hours to heat up to boiling if all cooling capacity is lost. There is ample time to effect repairs to the cooling system or arrange alternate cooling should adequate cooling capacity be lost. The amount of time before the pool begins to boil is dependent on both the heat load and the initial pool temperature. The heat loads as determined by the Standard Review Plan, for conservatism, were used for this analysis. With three pump-cooler configurations in operation with maximum heat load, prior to loss of forced cooling, the time to adiabatically heat up to boiling from an operating temperature of 150°F is shown in Table 6.3-1. For the normal heat load case with any two pump-cooler configurations in operation prior to loss of forced cooling, the time to adiabatically heat up to boiling from an operating temperature of 140°F is shown in Table 6.3-2.

TABLE 6.3-1

Time to Boiling in the Unit 3 Spent Fuel Pool
Three pump-cooler configurations in operation prior to loss of F.C.

Heat Load (10^6 BTU/hr)	Initial Pool Temperature (°F)	Heatup time (hrs)
30.8	150	5

TABLE 6.3-2

Time to Boiling in the Unit 3 Spent Fuel Pool
Any two pump-cooler configurations in operation prior to loss of F.C.

Heat Load (10^6 BTU/hr)	Initial Pool Temperature (°F)	Heatup time (hrs)
12.6	140	15

After boiling starts with the maximum heat load and no forced cooling, the required makeup rate will be less than 70 GPM.

Duke Power Company

Oconee Nuclear Station

Attachment 3

No Significant Hazards Consideration
Evaluation

10 CFR §50.91 requires that requests for amendment must be accompanied by an evaluation of the hazards considerations involved. Such evaluation is to focus on the three standards set forth in 10 CFR §50.91(b) as quoted below:

> The Commission may make a final determination pursuant to the procedures in §50.91 that a proposed amendment to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility involves no significant hazards consideration, unless it finds that operation of the facility in accordance with the proposed amendment would:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

As set forth more fully below, Duke Power Company ("Duke") submits that the activities associated with this amendment request are outside the standards set forth in 10 CFR §50.91(b) and, accordingly, a no significant hazards consideration finding is warranted. To put the matter in perspective, necessary background is first provided; thereafter, a discussion of each of the Significant Safety Hazard Considerations is provided.

The Oconee Nuclear Station was designed and constructed with two spent fuel storage pools--one associated with Units 1 and 2 and one with Unit 3. The design capacities of the pools were 336 spaces (1 2/3 core) and 216 spaces (1 1/3 core), respectively. The Oconee Final Safety Analysis Report addresses the safety implications of such pools to include relevant parameters associated with criticality, structural integrity, and cooling (Safety Evaluation, Docket Nos. 50-269/270/287). The evaluation found the environmental and safety impacts of such storage to be acceptable.

In 1975 it was deemed prudent to increase the storage capacity at the Oconee site. The Unit 1 and 2 pool contained spent fuel from the initial Unit 1 refueling in 1974. The Unit 3 pool did not contain any spent fuel. Thus, it was decided to increase the capacity of the Unit 3 pool. A request to amend the Unit 3 Operating License, DPR-55, was submitted on September 12, 1975 and was approved, as Amendment No. 17, on December 22, 1975. Approval of the amendment entailed detailed review and analyses of all relevant storage parameters and potential accidents. The analyses resulted in findings that reflected that the environmental and safety impacts were negligible; reference the Safety Evaluation issued December 22, 1975 in support of increasing the Unit 3 spent fuel pool from 216 to 474 fuel assemblies (including "failed fuel" locations).

The Safety Evaluation performed in support of the request to amend Unit 3 Operating License DPR-55 to allow reracking of the Unit 3 pool addressed the following areas:

1. Criticality analysis
2. Rack structural design (seismic design analysis)
3. Thermal consideration
4. Radiation levels
5. Accident consideration (fuel handling accident)
6. Spent fuel cask drop accident[1]

It was determined that the proposed modifications to the Oconee Unit 3 spent fuel pool would be acceptable because (1) the design would preclude criticality for any moderating condition, (2) the rack structural design adequately provided for seismic conditions, (3) the existing spent fuel pool cooling system was determined to have sufficient capacity to provide adequate cooling for the increased heat load, and (4) the increased radiation doses both onsite and off-site would be negligible.

It was considered at that time that the resulting combined onsite capacity (810 locations) would be sufficient to store spent fuel until such time as shipment to the Allied General Nuclear Services reprocessing plant could begin.

On April 17, 1977, President Carter issued a policy statement on commercial reprocessing of spent nuclear fuel which effectively eliminated reprocessing as part of the relatively near term nuclear fuel cycle. On October 18, 1977, the GESMO proceedings were deferred indefinitely. The combined effect of this national policy was to leave operating nuclear plants, like Oconee, without a repository for the spent fuel previously generated or being generated, other than expanding the spent fuel storage pool. Thus, Duke was forced to do additional reracking of the Oconee pools to further increase its storage capacity.

By letter dated February 2, 1979, Duke requested authorization to expand the capacity of the Unit 1 and 2 pool utilizing high-capacity non-poison racks. The expansion of the Unit 1 and 2 pool capacity as approved allowed the storage of up to 750 assemblies in that pool and 1224 onsite (including "failed fuel" locations). Again a detailed analysis of identical relevant parameters regarding virtually the same mechanistic conditions associated with the Unit 3 spent fuel pool enlargement preceded approval of the application. Findings in the Safety Evaluation issued June 14, 1979 again reflected that the environmental and safety impacts were negligible.

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[1]The Staff review of the spent fuel cask drop was not completed at this time and was scheduled for completion in early 1976. It was determined by the Staff that a completed spent fuel cask drop accident analysis was not a prerequisite for approval of the proposed modification.

By letter dated July 25, 1980, as supplemented by seven other submittals, Duke requested authorization to use Westinghouse designed/constructed poison racks in the Oconee 1 and 2 pool. By letter dated December 24, 1980, the NRC issued Amendments 90, 90, and 87, which authorized the reracking. Completion of the reracking increased the Unit 1 and 2 pool spent fuel storage capacity from 750 to a maximum of 1312 fuel assemblies. Once again detailed review and analyses of the same relevant parameters involving virtually the same mechanistic conditions associated with the two prior rerackings resulted in findings which reflected that the environmental and safety impacts were negligible.

The current fuel storage capacity at Oconee, therefore, consists of 1312 storage spaces in the Oconee 1 and 2 shared pool and 474 spaces in the Oconee 3 pool. With this application Duke Power is requesting approval to use, once again, Westinghouse designed/constructed poison racks to increase the Oconee 3 storage capacity to 825 spaces. This modification would extend the Oconee fuel storage capability from the current September 1988 date to October 1991. With the proposed reracking the full core reserve capability would be extended from January 1988 to March 1990.

The increase in Oconee 3 storage capacity would be accomplished by replacing the existing 14.09 inch center-to-center high density racks with 10.60 inch center-to-center neutron absorbing racks. These racks are of the same basic design as those currently utilized in the Oconee 1 and 2 storage pool.

Duke's analysis of the proposed amendment request is set forth in Attachment 2. Such analysis, as noted in the cover letter to this amendment request, addresses all of the areas addressed in the Staff's December 24, 1980 evaluation of the reracking of Oconee Unit 1 and 2 shared spent fuel pool with neutron absorbing spent fuel racks and addresses them in the same manner. Duke would note that the areas discussed are also identical to the areas addressed in the first three rerackings as well as the areas addressed in the over twenty reracking SERs (including PWR poison rerackings) that Duke has examined.

The following evaluation demonstrates by reference to the analysis contained in Attachment 2 that not one of the three significant safety hazards consideration standards are met. Each of the three standards is discussed below.

First Standard

> Involve a significant increase in the probability or consequence of an accident previously evaluated.

The analysis of this proposed reracking has been accomplished using current NRC Staff accepted Codes and Standards as specified in Section 2.1.2 of Attachment 2. The results of the analysis meet the specified acceptance criteria set forth in these standards. In addition, Duke has reviewed NRC Staff Safety Evaluation Reports for prior PWR rerackings involving poison racks to ensure that there are no identified concerns not fully addressed in this submittal.

From our analyses and SER reviews Duke has identified the following potential accident scenarios: 1) spent fuel cask drop; 2) loss of spent fuel pool forced cooling; 3) seismic event; 4) spent fuel assembly drop; and 5) construction accident. The probability of any of the first four accidents is not affected by the racks themselves; thus, reracking cannot increase the probability of these accidents. As for the construction accident, the proposed Oconee 3 pool reracking will not involve an increase in probability of any previously evaluated construction accident as accepted construction standards and procedures will be employed as described in Sections 4.0 and 6.1 of Attachment 2 of this submittal. Since there will be no fuel assemblies in the fuel pool during rack installation, the probability of some types of postulated construction accidents has actually decreased.

The consequences of the 1) spent fuel cask drop accident have been evaluated as described in Section 6.2 of Attachment 2. By limiting the age of fuel stored in the first 31 rows to not less than 70 days prior to any cask movement, the consequences of this type accident would be less than with the present racks as described in the Oconee FSAR Section 15.11.2.2. Thus, the consequences of this type accident will not be significantly increased from previous accident analyses.

The consequences of the 2) loss of spent fuel pool forced cooling accident have been evaluated and are described in Section 6.3 of Attachment 2. As indicated in Tables 6.3-1 and 6.3-2 of Attachment 2, there is ample time to effect repairs to the cooling system or to establish a makeup flow, and since the required makeup flow is less than the 70 gpm rate accepted by the NRC Staff for the Oconee 1 and 2 pool, the consequences of this type accident will not be significantly increased from previously evaluated accidents by this proposed reracking.

The consequences of a 3) seismic event have been evaluated and are described in Section 2.3.1 of Attachment 2. The racks were evaluated against the appropriate NRC Standard described in Section 2.1.2. The results of the seismic and structural analysis show that the proposed racks meet all of the NRC structural acceptance criteria and are consistent with results found acceptable by the NRC Staff in all previous poison rerack SERs including Oconee 1 and 2. Thus, the consequences of seismic events will not significantly increase from previously evaluated seismic events.

The consequences of a 4) spent fuel assembly drop accident are described in Section 2.3.1.5 of Attachment 2. The radiological consequences of this type accident are bounded by the cask drop accident and K_{eff} is shown to be always less than the NRC acceptance criteria of 0.95 and not significantly different from the margin to criticality found in the December 22, 1975 SER for the previous Oconee 3 rerack. Thus, the consequences of this type accident will not be significantly increased from previously evaluated spent fuel assembly drop accidents.

The consequences of a 5) construction accident are described in Section 6.1 of Attachment 2. Since there will be no fuel assemblies in the fuel pool during rack installation, there would be no radiological consequence of any construction accident. Thus, using accepted construction practices as described in Section 4.0 of Attachment 2 the consequences of a construction accident would be less than construction accidents previously evaluated by the NRC Staff.

Thus, it is shown that the proposed Oconee 3 spent fuel pool rerack 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.

Duke has evaluated the proposed reracking 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 Plans, and appropriate Industry Codes and Standards as described in Section 2.1.2 of Attachment 2. In addition, Duke has reviewed previous NRC Safety Evaluation Reports for poison rerack applications. In Duke's analysis and review of NRC evaluations and Industry Standards and Codes, Duke finds that the proposed reracking does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated including those on the Oconee 3 Docket.

Third Standard

> Involve a significant reduction in a margin of safety.

The issue of margin of safety when applied to a reracking modification will need to address the following areas (as established by the NRC Staff Safety Evaluation review process):

1. Nuclear criticality considerations
2. Thermal-hydraulic considerations
3. 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 spent fuel pool is to be less than or equal to 0.95, including all uncertainties, under all conditions. For the proposed modification, the criticality analysis, as discussed in Section 2.3.2 of Attachment 2, is exactly the same as that which was approved by the NRC Staff (SER issued December 24, 1980) for the Unit 1 and 2 shared pool reracking modification. The exact same codes, techniques, and assumptions were made. All aspects of the bases of the SER conclusions are covered in the identical manner.

The methods utilized in the analysis conform with ANSI N18.2-1973 "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", Section 5.7, Fuel Handling System; ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations", Section 5.1.12; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety", NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage"; and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

The results of this analysis indicate that K_{eff} is always less than 0.95 including uncertainties at a 95/95 probability/confidence level. Thus meeting the acceptance criteria for criticality, the proposed rerack 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 2.3.3 of Attachment 2. 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 reracking will allow an increase in the heat load in the Oconee 3 spent fuel pool. The evaluation in Section 3 of Attachment 2 shows that a third spent fuel cooling train will be added prior to putting more than the currently authorized 474 Fuel Assemblies in the spent fuel pool. The addition of the third cooling train will ensure that the pool temperature margins of safety of 150°F and 205°F described in Section 9.1.3 of the Oconee FSAR are maintained. Thus, there is no significant reduction in the margin of safety from a thermal-hydraulic standpoint or from a spent fuel cooling standpoint.

The mechanical, material, and structural considerations of the proposed rerack are described in Sections 2.1, 2.2, and 2.3 of Attachment 2. As described in Section 2.1, the racks are designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979. The racks are designed to Seismic Category 1 requirements and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support Structures. In addition, the racks are designed to withstand the loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane. The materials utilized are described in Sections 2.2 and 2.3.4 and are compatible with the spent fuel pool and the spent fuel assemblies. The structural considerations of the racks are described in Section 2.3 and show that the margin of safety against tilting is greater than 100, that the racks do not impact each other nor impact the pool walls, and that sufficient clearance is provided to prevent the racks from sliding into pool floor obstructions. Thus, the margin of safety is not sigsificantly reduced by the proposed rerack.

Thus, it has been shown that the proposed Oconee 3 Spent Fuel Pool does <u>not</u>:

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

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

As such, Duke has determined and submits that the proposed rerack described herein does <u>not</u> involve a significant safety hazard.