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(2-76)

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**DESCRIPTION**

Ltr notarized 4-16-76.......trans the follow:

**ENCLOSURE**

Amdt to OL/Change to Tech Specs: Consisting of changing relating to dry storage of new fuel assemblies in fuel storage racks located in Unit 3 spent fuel pool..............

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NRC FORM 195 (2-76)

# DUKE POWER COMPANY

POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083



April 16, 1976

Mr. Benard C. Rusche
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. A. Purple, Chief
Operating Reactors Branch No. 1

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

In accordance with Section 50.90 of 10 CFR 50, Duke Power Company hereby requests a change in the Oconee Nuclear Station Technical Specifications. The proposed change relates to the dry storage of new fuel assemblies in fuel storage racks located in Unit 3 spent fuel pool.

At present, the Units 1 - 2 spent fuel pool contains two batches of spent fuel assemblies and one batch of new fuel assemblies. With this inventory of fuel assemblies, the Units 1 - 2 spent fuel pool is not capable of receiving a full core discharge of fuel assemblies from either Unit 1 or Unit 2 reactors. Therefore, to facilitate the possible unloading of either the Unit 1 core or the Unit 2 core, the new fuel assemblies stored in the Units 1 - 2 spent fuel pool are to be relocated. At present, four modules of the new fuel storage rack design have been installed in the Unit 3 spent fuel pool and are capable of storing the new fuel assemblies. However, it is important that the new fuel assemblies be stored dry in the Unit 3 spent fuel pool at this time to enable the installation of the remaining rack modules.

The "Oconee Nuclear Station Unit 3 Spent Fuel Storage Facility Modification Safety Analysis Report", submitted to you with my letter of September 12, 1975, and subsequently revised with my letter of November 12, 1975, included detailed description of the design of the new fuel storage rack, structural and seismic analysis, criticality analysis, and the safety analysis of the utilization of these racks to store new and spent fuel assemblies. The NRC approval for the utilization of these racks to store new and spent fuel assemblies was granted by your letter of December 22, 1975. Although six rack modules remain to be installed, the structural and seismic, the criticality, and the safety analyses, described in the aforementioned safety analysis report, continue to be valid for the utilization of the racks in the present arrangement. Additional detailed criticality analysis has been performed to support the dry storage of new fuel assemblies in the Unit 3 spent fuel pool. This criticality analysis has been performed by Combustion Engineering, Incorporated, supplier of the Unit 3 fuel storage racks. This analysis shows that an adequate degree of subcriticality will be maintained during all credible storage conditions. Attachment 1 is a summary of the criticality analysis including the assumptions employed.

Attachment 2 contains replacement pages 4.1-10, 5.4-1, and 5.4-1a of the Oconee Nuclear Station Technical Specifications with the proposed changes identified by vertical lines.

We request that approval of the Technical Specifications change be granted as expeditiously as possible and no later than June 1, 1976.

Forty copies of this request, including three signed originals, are enclosed.

Very truly yours,

s/William O. Parker, Jr.

William O. Parker, Jr.

PMA:mmb

Attachments

WILLIAM O. PARKER, JR., 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/William O. Parker, Jr.
William O. Parker, Jr., Vice President

ATTEST:

s/John C. Goodman, Jr.
John C. Goodman, Jr.
Assistant Secretary
(Seal)

Subscribed and sworn to before me this 16th day of April 1976.

s/Edna B. Farmer
Notary Public
(Notarial Seal)

My Commission Expires:

October 24, 1977

ATTACHMENT 1

# CRITICALITY EVALUATION FOR DRY STORAGE

# OF FRESH FUEL ASSEMBLIES

# IN OCONEE UNIT 3 SPENT FUEL POOL

# CRITICALITY EVALUATION FOR DRY STORAGE OF FRESH FUEL ASSEMBLIES IN OCONEE UNIT 3 SPENT FUEL POOL

## I. INTRODUCTION AND SUMMARY

A criticality analysis is provided herein to support the proposed dry storage of fresh fuel assemblies of enrichment up to 2.9 weight percent U-235 in the high capacity fuel storage rack modules located in Oconee Unit 3 spent fuel pool by demonstrating that the multiplication factor of the array is less than or equal to the design limit multiplication factors of 0.95 under flooded conditions and 0.98 under optimum moderation conditions.

Placement of fuel assemblies in each storage location and a checkerboard loading scheme (only diagonally adjacent storage locations being occupied by fuel assemblies) are examined. Results of the analysis show that under fully flooded or uniformly dispersed aqueous form conditions either fuel loading scheme results in multiplication factors less than 0.88. As expected, the checkerboard loading pattern results in lower calculated multiplication factors than the fully loaded pattern and provides margins of 0.12 and 0.36 in units of k at the fully flooded and mist conditions, respectively. The analyses for the fully loaded rack conditions demonstrate that the inadvertent misloading of a single fuel assembly in an otherwise checkerboard array does not lead to a violation of subcriticality margins.

Potential non-uniform flooding conditions have also been examined. Specifically, it is hypothesized that the storage cells were filled with full density water and that the space between storage cells was filled with lower density water. Under this hypothetical condition also, the checkerboard loading pattern was found to be acceptable.

## II. DESIGN BASES AND INPUT PARAMETERS

A criticality analysis is performed to demonstrate that the effective multiplication factor of the normally dry fuel storage rack, when loaded with fresh fuel of the highest anticipated enrichment, will not exceed:

(1) 0.95 when flooded with pure water, and (2) 0.98 assuming optimum moderation (aqueous foam condition). These criteria are consistent with the applicable industrial standard, ANSI N18.2[1].

The maximum enrichment of the fuel assemblies to be stored in the normally dry rack is assumed to be less than or equal to 2.9 w/o $U^{235}$. Relevant physical parameters of the fuel assemblies employed in the analysis are the nominal design values. Where ranges of parameters are shown, extremum values were chosen such that the predicted multiplication factor of the storage rack is a maximum. The inherent neutron-absorbing effect of the stainless steel storage box wall structure is explicitly treated in the analysis. Credit has not been taken for neutron absorption of the assembly grid spacers and end fittings, nor for neutron absorption by structural steel components of the storage rack other than the individual storage box wall structure.

Storage cells and modules are shown in Reference 2. The analysis is performed assuming that the storage cells consist of rectangular boxes, with a nominal inside dimension of 9.375 inches, constructed of 0.25 inch thick type 304 stainless steel (the guide funnel and end casting are neglected). These boxes are welded to structual components to form storage modules with a nominal center-to-center distance between adjacent boxes of 14.09 inches. Two modules of 6 by 8 storage cells each are welded together to form a regular 8 by 12 array of storage cells. To account for manufacturing tolerances, it is assumed that the dimensions (within tolerance) of the storage cells and modules are such that the predicted multiplication factor is a maximum. Hence, results of the analysis presented in this report are based on an assumed: (1) minimum cell center-to-center spacing of 13.965 in., (2) minimum box wall thickness of 0.24 in., and (3) maximum box inside dimension of 9.9375 in. To conservatively represent neutron scattering by materials surrounding the storage rack, it has been assumed that the array is bounded by a three foot thick concrete wall spaced one foot from the edge of the storage array on all six sides.

## III. ANALYTICAL METHODS

### A. General

In order to more accurately predict the multiplication factor of the storage arrays, reliable calculation of the spatial flux distribution, especially in the neutron absorbing steel regions, is essential. For this reason, one and two dimensional transport calculations for the storage rack are employed. In the two dimensional transport calculations, each component of the fuel storage location "cell" is explicitly represented. Thus, in the normal storage cell calculation, the fuel assembly, the water channel between the fuel assembly and the box walls, the steel box, and one half of the water gap between adjacent storage locations are represented as separate regions. The fuel assembly itself is represented as a 15 x 15 array of cells containing moderator and either fuel pins, guide tubes or instrument tubes. Four neutron group cross sections are generated for each fuel assembly cell and for each component of the storage cell with special attention given to the effect of adjoining regions on the spatial thermal neutron spectrum and hence broad group thermal cross sections of each separate region of the storage cell. Flux-volume weighted cross sections, extracted from the two dimensional transport calculations, are used in the one dimensional transport calculations as described below.

### B. Cross Section Generation

The CEPAK lattice program (Version 2.2 Mod 10) is used to calculate four neutron group cross sections. This program is the synthesis of a number of computer codes, many of which were developed at other laboratories, e.g., FORM[3], THERMOS[4], and CINDER[5]. These programs are interlinked in a consistent way with an extensive library of differential neutron groups between 0 and 10 Mev. Neutron leakage in a single Fourier mode is represented by a B-1 approximation to transport theory throughout this entire range. Resonance shielding is determined analytically. The effective fuel temperature is incorporated into the calculational model by means of the Hellstrand correlation renormalized to a gold resonance integral of 1565 barns. This correlation is a semi-empirical fit to experimental data for both metal and oxide uranium rods. The Hellstrand correlation[6] is employed for U-238, with appropriate adjustments guided by Monte Carlo calculations of resonance capture in U-238 so as to provide agreement with selected measurements of the conversion ratio. Plutonium resonance integrals are determined from an intermediate resonance formulation using equivalence relationships for the lattice representation[7]. The Dancoff factor D, which is a measure of the shielding of a fuel rod resulting from the presence of neighboring fuel rods, is calculated by the Fukai method[8] for a uniform lattice. This method carries out the numerical integrations necessary for the computation of the moderator and clad transmission probabilities. Vacancies in the lattice are treated by an approximation used successfully by Hicks[9] which apportions the uniform lattice Dancoff correction C, (C = 1 - D), equally among the nearest neighbors.

The data base for both fast and thermal neutron cross sections for this version of the CEPAK program is derived from several sources, mainly ENDF/B-II, BNL-235, and early Bettis libraries. This data base gives good agreement with measured data from critical experiments and operating reactors. The standard multigroup cross section library employed in the CEPAK lattice program for SS-304 has a macroscopic 2200 m/s absorption cross section of 0.2597 $cm^{-1}$. The use of ENDF/B-4 2200 m/s absorption cross sections would yield a larger 2200 m/s macroscopic cross section by approximately 3.7% with a variation of approximately 1% due to typical variations in nuclide composition and density of the type 304 alloy. Thus the 2200 m/s value of the absorption cross section derived from CEPAK should yield a more conservative thermal absorption rate in SS-304 than one derived from ENDF/B-4 data sources.

The fuel assembly region of a storage cell is represented by a 15 x 15 array of fuel assembly cells having a basic pitch of 0.568 inches and has an overall square dimension of 8.52 inches. Microscopic cross sections for nuclides in the fuel assembly cells as well as those exterior to the fuel assembly but within the outer boundary of the stainless steel box are averaged over the multigroup spectrum calculated by the FORM portion of the CEPAK lattice program for a homogenized representation of the fuel assembly. The broad group thermal cross sections are obtained from the one-dimensional THERMOS portion of CEPAK for each type of fuel assembly cell; control rod guide tube and instrument cells employ an explicit representation of moderator and structure within thecell and a homogenized fuel pin cell environment. Four broad neutron group (3 fast and 1 thermal) microscopic cross section edits are obtained from the CEPAK lattice program. Heterogeneous fast fission effects are included in the top broad group cross sections by applying correction factors derived from an auxiliary two-dimensional integral transport calculation that employs the collision probability technique to compute sub-region dependent reaction rates in an explicit geometric representation of the fuel rods and associated structure of a fuel assembly. The correction factors are the relative flux ratios for the fuel, clad, and moderator within a fuel rod cell. The 3 fast broad group cross sections for the moderator region between storage boxes are obtained from a uniform moderator medium CEPAK calculation using water of an appropriate density as the moderating material. The thermal cross sections for the water and steel regions are derived from slab geometry THERMOS calculations with an appropriate fuel assembly environment.

C. Two Dimensional Transport Calculations

The two dimensional, discrete ordinates transport code DOT-2W[10] (Version 1.0 MOD 1 - May 7, 1973) is used to determine the spatial flux solution and multiplication factor. An S-6 order of angular quadrature is used with a 1.0001 convergence factor (the ratio of successive eigenvalues for each outer iteration). In the fully loaded storage cell calculations, one quarter of an assembly is represented with one mesh interval for each fuel assembly cell; the surrounding water channel, steel, and water gap regions are calculated with 2, 4 and 6 intervals, respectively. Thus the X-Y representation of the fully loaded storage cell is a 20 x 20 mesh interval problem. The same general principles are followed in the representation of the checkerboard loading scheme. In this model one quarter of each storage cell of a cluster of four storage cells is represented. The X-Y DOT representation of the checkerboard loaded storage array is a 40 x 40 mesh interval problem.

D. One Dimensional Transport Calculations

Non-leakage probability values of the storage rack are obtained using the one dimensional transport code ANISN[11] (Version 1.0, MOD. 0) as described below. These calculations are performed using 4 energy group modifed Po cross sections and an $S_8$ quadrature set. Three regions are represented explicitly in these calculations: (1) a fuel assembly and storage rack region with flux-volume weighted cross sections obtained from the two dimensional transport (DOT) calculations, (2) an aqueous foam region, and (3) a concrete region. The latter regions are represented using cross sections obtained using the CEPAK code. The ANISN calculations are performed at several water densities of interest for: (1) an infinite storage rack, (2) a rack 8 storage cells wide, (3) a rack 12 cells long, and (4) a rack 144 inches high. The latter three calculations are performed assuming one foot of low density water and three feet of concrete surrounding the rack. The non-leakage probability is defined by the following equation.

$$P_{NL} = \frac{\kappa(\text{width}) \times \kappa(\text{length}) \times \kappa(\text{height})}{(\kappa\infty)^3}$$

where: 1. $P_{NL}$ is the non-leakage probability

2. $\kappa$(width), $\kappa$(length), $\kappa$(height) are the computed multiplication factors assuming the storage rack is of infinite extent in two directions and finite in the third dimension, i.e., length, width or height.

3. $\kappa\infty$ is the computed multiplication factor assuming the rack extends infinitely in all directions.

## IV. RESULTS

Past experience from criticality evaluations for dry fuel storage racks has shown that the multiplication factor varies with the assumed density of water dispersed uniformly througout an infinite array of fuel storage cells in the following fashion. As the water density is reduced below the value at 68°F, the multiplication factor decreases in a monotonic manner to a water density in the range of 0.5 gm/cc; as the water density is reduced to zero, the multiplication factor passes through a maximum. The maximum value of the multiplication factor at both the full and reduced water density conditions is a function of the fuel enrichment, size of the fuel assembly, lattice pitch of the fuel assembly storage array, and the amount and distribution of parasitic structural material in the storage rack. For the conditions where two 6 x 8 HI-CAP type fuel storage modules are combined to form an 8 x 12 array of fuel storage locations, the lattice spacing and array size are fixed; the only remaining variable is the fuel loading configuration since the limiting enrichment is set at 2.9 w/o.

Figure 1 summarizes the results of analyses for an array of fuel storage locations which are of infinite extent in all directions. The data points at a relative water density of 1.00 correspond to complete immersion of the rack in water at 68$^{o}$F for three cases - all storage locations filled by fuel assemblies having enrichments of 3.5 w/o and 2.9 w.o, and a checkerboard array of 2.9 w/o enriched fuel assemblies. The calculated multiplication factors are 0.9070, 0.8711, and 0.8233, respectively, assuming concurrent adverse dimensional tolerances as specified in Section II. These data points are of interest for comparing the change in multiplication factor with the changes in enrichment and fuel arrangement relative to the licensed conditions when the fuel storage modules are employed in the Oconee Nuclear Station, Unit 3 spent fuel pool.

In the event that the fuel storage array could be exposed to a sufficiently large volume of water from fire fighting apparatus, pipe breaks, etc., such that the funnel at the top of each storage location would divert most of the water to the interior of the storage box, it is postulated that the most adverse condition would be a complete flooding of the interior of each fuel storage box, with a relatively low (0.02 gm/cc) water density between storage boxes. Analyses for this postulated condition indicate that for the checkerboard arrangement of fuel assemblies, the multiplication factor for the infinite array increases from 0.8233 (uniformly flooded) to 0.8404. The calculated multiplication factor of 0.8404 corresponding to this hypothetical condition shows a significant margin to the design limit of 0.98.

Results are shown in Figure 1 at reduced water density conditions for both the checkerboard and fully loaded rack conditions. For each fuel loading pattern and relative water density, an upper and lower bound in the calculated multiplication factor is shown. The range of values corresponds to changes in the four group macroscopic cross sections due to large variations in the multigroup spectrum. These variations are induced, via the buckling parameter, to examine the sensitivity of the results to the slowing-down spectrum in the fuel and bulk moderator regions. Values of the energy and spatial dependent neutron leakage inferred from the two dimensional transport calculations lie within the band of assumed input bucklings.

The multiplication factors shown in Figure 1 for reduced water density conditions show a margin to the design basis of 0.98 in excess of either 0.12 units of k for an infinite array of storage cells, each of which contains one fuel assembly, or 0.36 units of k for an infinite array of storage cells containing one fuel assembly in every other location (checkerboard array).

Figure 2 shows a plot of the non-leakage probability for the finite checkerboard array (8 x 12) of fuel assemblies at the reduced water density conditions. Figure 3 shows the effective multiplication factor (product of infinite multiplication factor and non-leakage probability) for the checkerboard array of 2.9 w/o fuel assemblies as a function of water density. For this finite checkerboard array of fuel assemblies at the optimum moderation conditions, a margin of 0.49 units of k exists relative to the design basis value of 0.98.

REFERENCES:

1. ANSI-N18.2-1973, "Americal Nuclear Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, August 6, 1973.

2. "Oconee Nuclear Station Unit 3 Spent Fuel Storage Facility Modification Safety Analysis Report," submitted to the NRC by Duke Power Company, letter of W. O. Parker, Jr. dated September 12, 1975.

3. FORM - A fourier Transform Fast Spectrum Code for the IBM-7090, McGoff, D. J., NAA-SR-Memo 5766 (September 1960).

4. THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations, Honeck, H., BNL-5816 (July 1961).

5. CINDER - A One-Point Depletion and Fission Product Program, England, T. R., WAPD-TM-334 (Revised June 1964).

6. Measurement of Resonance Integral, Hellstrand, Proceedings of National Topical Meeting of the ANS, San Diego, February 7-9, 1966, The M.I.T. Press.

7. "An Equivalence Formulation for Absorption in Plutonium," S. Borrensen and R. Goldstein, Trans. Am. Nuclear Society, 15, 296 (1972).

8. Y. Fukai, J. Nuclear Energy, 22, 355, 1968.

9. R. Alpiar, METHUSELAH I., AEEW-R-135, 1964.

10. R. G. Sottesy, R. L. Disney, A Collier, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code," WANL-TME-1982.

11. W. W. Engle, Jr., "A Users Manual for ANISN," K-1693, March 30, 1967.

Figure 1. Infinite Multiplication Factor vs Water Density

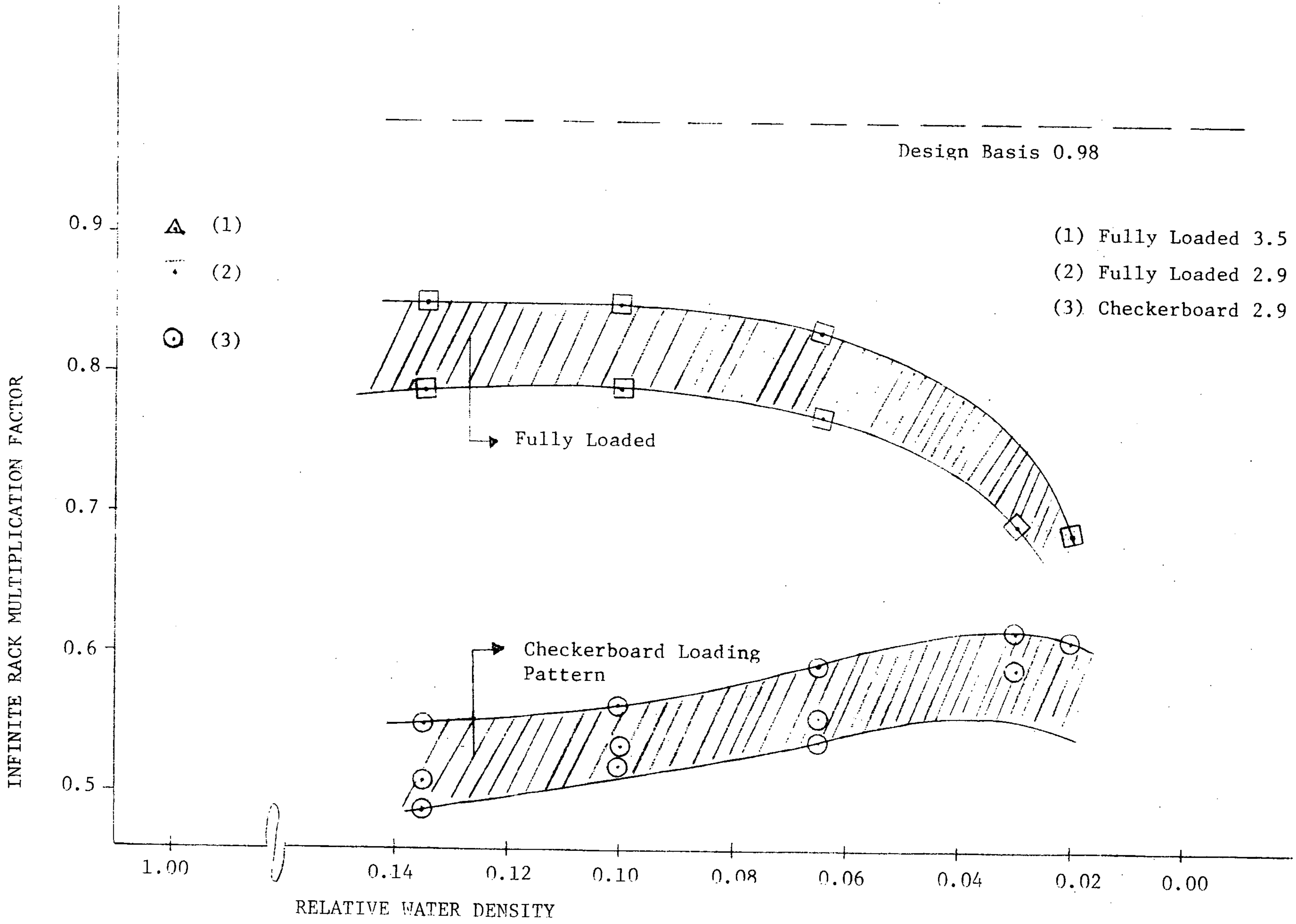


Figure 2.  Non-Leakage Probability of Checkerboard Array vs Water Density

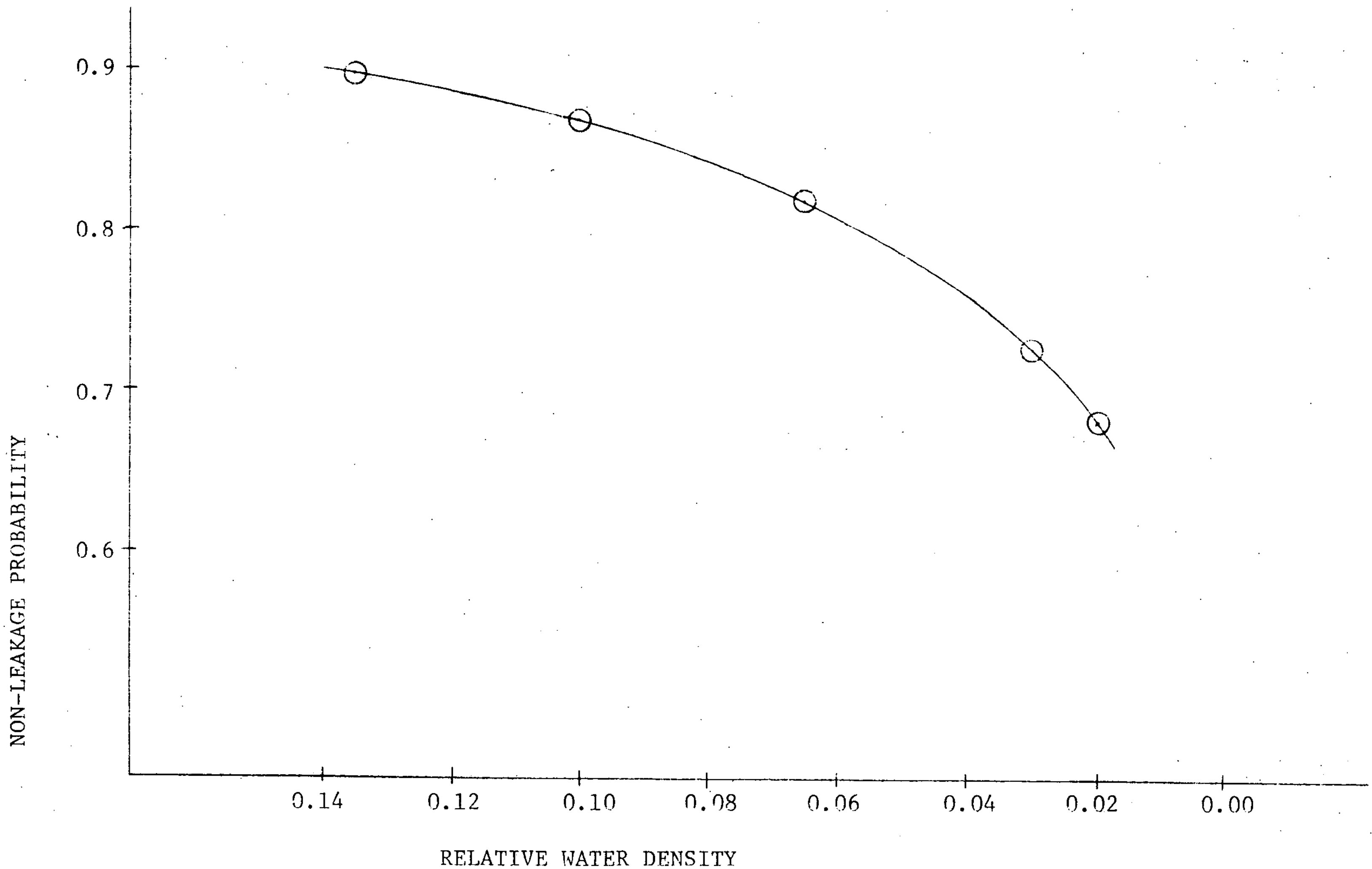
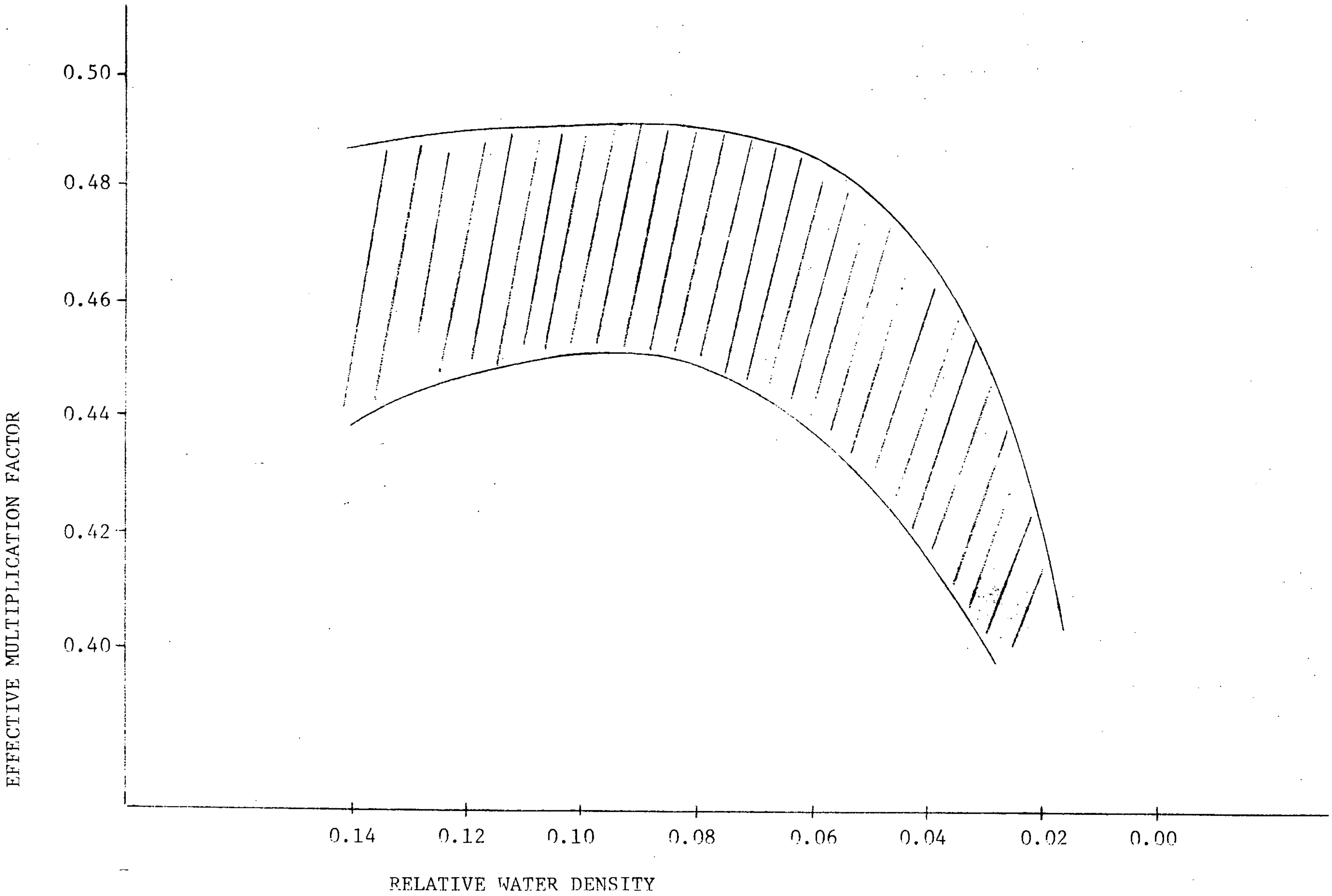


Figure 3. Effective Multiplication Factor of Checkerboard Array of Fuel Assemblies vs Water Density



ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

OCONEE NUCLEAR STATION

TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

| Item | Check | Frequency |
|---|---|---|
| 1. Reactor Coolant | a. Gamma Isotopic Analysis | a. Monthly* |
| | b. Radiochemical Analysis for Sr 89, 90 | b. Monthly* |
| | c. Tritium | c. Monthly* |
| | d. Gross Beta & Gamma Activity (1) | d. 5 times/week* |
| | e. Chemistry (Cl, F and O2) | e. 5 times/week* |
| | f. Boron Concentration | f. 2 times/week** |
| | g. Gross Alpha Activity | g. Monthly* |
| | h. $\overline{E}$ Determination (2) | h. Semi-annually |
| 2. Borated Water Storage Tank Water Sample | Boron Concentration | Weekly* and after each makeup |
| 3. Core Flooding Tank | Boron Concentration | Monthly* and after each makeup |
| 4. Spent Fuel Pool Water Sample | Boron Concentration | Monthly*** and after each makeup |
| 5. Secondary Coolant | a. Gross Beta & Gamma Activity | a. Weekly* |
| | b. Iodine Analysis (3) | |
| 6. Concentrated Boric Acid Tank | Boron Concentration | Twice weekly* |

*Not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency.
**Applicable only when fuel is in the reactor.
***Applicable only when fuel is in wet storage in the spent fuel pool.

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 21 inches in both directions. This spacing is sufficient to maintain a K effective of less than 0.9 when flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent $U^{235}$.

In the spent fuel pool serving Unit 3, the fuel assemblies are stored in racks consisting of stainless steel cavities which maintain a minimum edge-to-edge spacing of 3.95 inches between adjacent fuel assemblies. The neutron poisoning effect of the storage cavity material combined with the minimum 3.95 inches edge-to-edge spacing between adjacent fuel assemblies is sufficient to maintain a K effective of less than 0.95 when flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent $U^{235}$ or the equivalent.

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.

5.4.1.4 New fuel of enrichment not exceeding 2.9 weight percent U-235 or the equivalent may be placed in dry storage in Unit 3 fuel storage racks in a checkerboard pattern, with fuel assemblies occupying only diagonally adjacent storage locations. This configuration is sufficient to ensure a K effective of less than 0.9 at all times.

5.4.2 Spent Fuel Storage

5.4.2.1 Irradiated fuel assemblies will be stored, prior to offsite 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 Unit 3 spent fuel pool to accommodate up to 474 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 Except as provided in Specification 5.4.1.4, 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 9.7