DUKE POWER COMPANY
OCONEE NUCLEAR STATION

REVISIONS TO JULY 1, 1980
POISON RERACK LICENSING SUBMITTAL
FOR UNITS 1 AND 2 SPENT FUEL POOL

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August 14, 1980

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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.

Because of structural symmetry of the cell assembly about the x and y axes, the x and y direction horizontal seismic events produce identical loads. Consequently, the margins of safety for the multi-direction (x and y directions simultaneously) seismic event is computed by multiplying the uni-direction loads by the square root of 2.

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

The margins of safety, due to the multi-direction shock for the leveling pad assembly, are the same as the uni-direction because maximum stresses, due to x-direction and y-direction seismic acceleration, do not occur at the same point. The multi-direction seismic event stresses for the weld of the leveling pad assembly are properly corrected.

The loads summarized in the seismic analysis section 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 lbs. 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 is 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 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 (3,000 lbs) impacts on the top of the rack. Calculations show that the impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cell funnels and the section of cell above the upper grid structure 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 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 6.3.

The second accident condition assumes that the fuel assembly 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 will not be perforated and the margin of safety is positive. Critically calculations show that $K_{eff} < 0.95$ and the criticality acceptance criteria is not violated for the Oconee posion 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 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 allowable 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.

The calculational method uncertainty is discussed in Section 2.3.2.4.

e. 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 self shielding is included as a bias in the reactivity calculations.

2.3.2.3 Postulated Accidents

Most accident conditions will not result in an increase 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 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 protection against 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 is 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, 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.

This "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 removed 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 designs.

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 multi-group cross-section sets are then used as input to KENO IV [3] which is a three-dimensional Monte Carlo 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. (See Table 2.3-1 for summary of 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:

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

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

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

$$K_{eff} = K_{nominal} + B_{mech} + B_{method} + B_{part} + [(KS_{nominal})^2 + (KS_{method})^2 + (KS_{mech})^2]^{1/2}$$

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 by 0.5 inches.

Substituting calculated values, the result is:

$K_{eff} = 0.9475$

Since this K_{eff} is less than 0.95 including uncertainties at 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 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 $K_{eff} \leq 0.95$.

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 and the flow velocities and fluid 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.

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 at least three separate flow openings as shown in Figure 2-7. The use of these multiple flow holes virtually eliminates the possibility that all flow into the inlet

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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 3.2.3.

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. Under normal conditions, with three pump-cooler configurations in operation, the pool temperature is maintained under 125°F as stated in the Oconee FSAR Section 9.4.2.1 by recirculating spent fuel cooling water from the spent fuel pool through the pumps and coolers and back into the pool. The pumps and coolers are arranged in parallel. The purification function is performed as described in the Oconee FSAR Section 9.4.2.1.1.

The heat loads shown in Tables 3.2.1 and 3.2.2 represent the largest heat loads expected in the spent fuel pool. The normal case assumes that Units 1 & 2 are refueled consecutively and the pool is filled with previous discharges except those spaces reserved for full core discharge. The full core discharge case assumes consecutive refueling followed by a full core discharge after a short period of operation. In this case all the spaces in the pool are filled. The calculations assume that 18 month cycles are used on both units.

Pool temperature is maintained below 150°F by operation of any two pump-cooler configurations for the normal heat load and by operation of all three pumps and coolers for the maximum heat load. Upon failure of one pump or cooler for either of these conditions sufficient cooling capacity remains to maintain bulk pool temperature below 205°F. An analysis of pool response to loss of all forced cooling is presented in Section 6.4 of this document.

3.3 <u>WATER QUALITY</u>

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.