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DOCKET NO. 50-336

ATTACHMENT 1

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

PROPOSED TECHNICAL SPECIFICATIONS

MAY, 1980

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,060 + 700/-0 cubic feet.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between assemblies to ensure a $k_{eff} < 0.95$ with the storage pool filled with unborated water. The maximum fuel assembly enrichment to be stored in the fuel pool will be 3.35 w/o.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22'6".

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in Section 5.1.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 5.8 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

DOCKET NO. 50-336

ATTACHMENT 2

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

CRITICALITY ANALYSIS FOR MILLSTONE UNIT NO. 2

SPENT FUEL STORAGE POOL

MAY, 1980

NUCLEAR CONSIDERATIONS

1.0 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 to take advantage of neutron absorption in water and stainless steel.

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.

1.1 NORMAL STORAGE

a. Th 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 14 X 14 Westinghouse fuel assembly is 3.35 w/o U-235 with no depletion or fission product buildup. The assembly is conservatively modeled with water replacing the assembly grid volume and no U-234 and U-236 in the fuel pellet.

b. 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. The nominal center-to-center spacing is 12.19 inches.

c. 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. Calculations show that the finite rack surrounded by a water reflector is less reactive than the nominal case infinite rack. Therefore, the nominal case of an infinite array of cells is a conservative assumption.

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

- stainless steel thickness
- can ID
- center-to-center spacing
- asymmetric assembly position

The calculational method uncertainty and bias is discussed in Section 1.3.

e. Credit is taken for the neutron absorption only in the full length stainless steel box wall structure.

1.2 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 assembly has more than eight inches of water separating it from the rest of the rack which precludes interaction).

However, accidents can be postulated which could increase the reactivity in the spent fuel storage pool such as an inadvertent drop of a fuel assembly between the outside periphery of the spent fuel rack and the spent fuel pool wall. The design of the Millstone Unit No. 2 spent fuel pool and associated equipment assures that the closest physically achievable approach of a fuel assembly to the side of the spent fuel rack will not increase the multiplication factor of the storage rack.

For fuel storage applications, water is usually present. However, accidental criticality when fuel assemblies are stored in the dry condition is also accounted for. For this case, possible sources of moderation, such as those that could arise during fire fighting operations, are included in the analysis.

This "optimum moderation" accident is not a problem in 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).

1.3 CRITICALITY ANALYSIS

The calculational 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 ensures 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 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 and 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.

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:

$$TU = [\ (ks)^2_{method} + (ks)^2_{nominal} + (ks)^2_{asym}]^{1/2}$$

where $(ks)_{method}$ is 0.013 as discussed above, $(ks)_{nominal}$ is the statistical uncertainty associated with the particular KENO calculation being used, and $(ks)_{asym}$ is the statistical uncertainty due to asymmetric assembly position

The reactivity effect of mechanical tolerances is ignored in this analysis due to the assumption of "worst-case" geometry for the base case K_{eff}. For Millstone Unit 2, the worst combination of mechanical tolerances results in a 0.24 inch can thickness and a spacing between cans of 2.625 inches. No bias or uncertainty is therefore applied due to mechanical tolerances.

Another center-to-center spacing reduction can be caused by asymmetric assembly position within the storage can. The analysis assumes that groups of four assemblies are located within the storage cans so that the spacing between them is a minimum. This results in a center-to-center spacing reduction of 0.54 inches and a reactivity increase of 0.0105 Δk. This is conservatively taken as a bias even though the asymmetric position of an assembly within the storage can will be a random event.

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 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".

1.4 RACK MODIFICATION

The spent fuel storage rack is described in Reference 7. The worst case geometry assumptions (used to eliminate the sensitivity to mechanical tolerances) are identical to those used in previous spent fuel storage pool criticality analyses.

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

$$K_{eff} = K_{nominal} + B_{method} + B_{asym} + [(ks)^2_{nominal} + (ks)^2_{method} + (ks)^2_{asym}]^{1/2}$$

where:

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

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

B_{asym} = bias to account for asymmetric assembly position

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

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

ks_{asym} = 95/95 uncertainty to account for asymmetric assembly position

Substituting calculated values, the result is:

$K_{eff} = .9456$

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

The tabular reactivity balance for the criticality analysis of the spent fuel storage rack is shown below. Calculated values of the multiplication factor labeled nominal, minimum and most adverse correspond to calculations performed assuming, respectively, nominal rack dimensions, the most adverse concurrent combination of dimensional tolerances, and the most adverse combination with the simultaneous displacement of the fuel assemblies into their most reactive positions within the cans.

	Nominal	Minimum	Most Adverse
Multiplication Factor for Spent Fuel Storage Rack	.875	.918	.946
Excess Margin	.075	.032	.004

1.5 ACCEPTANCE CRITERIA FOR CRITICALITY

For the purposes of this analysis, the acceptance criterion has been chosen to be $K_{eff} \leq 0.95$, including all uncertainties and under all conditions.

The results of the analysis demonstrate that K_{eff} for the Millstone Unit No. 2 spent fuel storage pool will meet the acceptance criterion with fuel enriched to 3.35 w/o U-235 for all normal and accident conditions.

REFERENCES

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3. L. M. Petrie and N. F.Cross, "KENO IV-An Improved Monte Carlo Criticality Program," ORNL-4938 (November 1975).

4. S. R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 2.35 wt % ^{235}U Enriched UO_2 Rods in Water With Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2438 (October 1977).

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6. J. T. Thomas, "Critical Three-Dimensional Arrays of U (93.2) - Metal Cylinders," Nuclear Science and Engineering, Volume 52, pages 350 - 359 (1973).

7. "Millstone Unit No. 2 Spent Fuel Pool Modifications", Docket 50-336, November 22, 1976.