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# CRITICALITY SAFETY ANALYSES
# FOR THREE MILE ISLAND UNIT 1
# FUEL STORAGE FACILITIES

Prepared for

GPU NUCLEAR

by

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## 1.0 INTRODUCTION

The fuel storage facilities at the Three Mile Island Unit 1 Nuclear Power Plant are currently licensed for 3.5% U-235 enriched fuel. There are four storage facilities involved as follows:

- o the new fuel receiving and storage vault,
- o the storage rack in the transfer canal,
- o the Pool A spent fuel storage pool, and
- o the Pool B spent fuel storage pool.

Each of these facilities are capable of safely storing fuel of enrichments greater than 3.5% within USNRC guidelines and the present analysis was undertaken to justify the criticality safety of an increase in Technical Specification limits on fuel enrichment. Results of the analysis demonstrate that the Pool B spent fuel storage racks are the most restrictive. When fully loaded with fuel of 4.3% enrichment and flooded with unborated water at the temperature of highest reactivity, the maximum reactivity (k-infinite) of the Pool B racks is less than the NRC limit of 0.95 including all known uncertainties. The new fuel storage vault (with certain restrictions), the canal racks, and the Pool A racks are capable of accommodating higher enrichments but are here evaluated at 4.3% enrichment as limited by the Pool B racks.

The nominal fuel assembly design is the standard Babcock & Wilcox 15x15 fuel assembly. Minor changes in fuel specifications may be accommodated by the use of equivalency factors on allowable enrichment as described in this report.

Acceptability of the new fuel storage vault for fuel of 4.3% enrichment requires a special restriction in order to meet the SRP 9.1.1 requirements under hypothetical conditions of low density "optimum" moderation. This restriction is that 12 storage locations in two rows must remain unused and empty of fuel or moderating material in order to provide necessary additional neutron leakage to maintain the reactivity at an acceptable level under the postulated accident conditions.

The racks in the transfer canal are identical to the Pool A racks in lattice spacing and therefore the criticality safety analyses for the Pool A racks also apply to the canal racks. References to the Pool A analyses in this report should be interpreted to also refer to the racks in the transfer canal. In practice, because of their smaller size, the canal racks would exhibit a slightly lower k-effective for the same fuel enrichment if neutron leakage were to be included.

Details of the analyses and results are presented herein.

## 2.0 SUMMARY

The criticality safety analyses of the fuel storage facilities at the Three Mile Island Unit 1 plant are summarized in Table 1 for fuel of 4.3% enrichment. As shown in this table, Pool B exhibits the highest - and limiting - reactivity. For Pool B at 120°C, the maximum k-infinite is conservatively estimated to be 0.949 under the statistical combination of all known calculational and mechanical uncertainties with a 95% probability at a 95% confidence level. The new fuel vault, and the Pool A racks (and canal racks) exhibit an even lower reactivity and, therefore, all storage facilities conform to the applicable NRC requirements. Credible abnormal or accident conditions will not result in exceeding the limiting reactivity specified in the NRC guidelines.

For the new fuel storage vault to be acceptable under optimum low moderator density conditions specified in SRP 9.1.1, twelve locations must remain empty of fuel or moderating material in order to provide necessary additional neutron leakage. These 12 locations, in two rows of 6 locations each, are shown as black bars on Figure 1. It is this configuration for which the new fuel vault calculations in Table 1 are applicable, with a maximum k-effective of 0.928 for the hypothetical optimum low moderator density accident case where the NRC limiting value is 0.98 (SRP 9.1.1). Under the fully flooded accident condition, where the NRC limit on k-infinite is 0.95, the new fuel vault will resemble the Pool A racks with a maximum k-infinite of 0.936.

Although the Pool B rack analysis considered the standard B&W 15x15 fuel assembly at 4.3% enrichment with typical manufacturing tolerances, possible future minor variations in fuel assembly design may be accommodated by adjusting the maximum allowable enrichment. The reactivity equivalence factors are:

- o 0.02 decrease in percent enrichment per 1% increase in $UO_2$ density from the nominal 10.225 g/cc,and
- o 0.035 decrease in percent enrichment per 1% increase in fuel pellet diameter from the nominal 0.369 inches.

Table 1 SUMMARY OF CRITICALITY SAFETY ANALYSES

| | New Fuel Vault | Pool A | Pool B |
|---|---|---|---|
| Calculational method | KENO[1] | CASMO-2E | CASMO-2E |
| Temperature of evaluation | 20°C | 20°C | 120°C |
| Enrichment, wt% U-235 | 4.3 | 4.3 | 4.3 |
| Nominal K-infinite | 0.9131[1] | 0.9308 | 0.9423 |
| Calculational bias | 0.0024 | 0.0013 | 0.0013 |
| Uncertainties | | | |
| Bias | ±0.0030 | ±0.0018 | ±0.0018 |
| Calculation statistics | ±0.0113 | NA | NA |
| Lattice Pitch | negligible | ±0.0005 | ±0.0021 |
| SS thickness | NA | NA | ±0.0012 |
| SS Box ID | NA | NA | ±0.0002 |
| Pellet diameter | ±0.0020 | ±0.0020 | ±0.0020 |
| Fuel Enrichment | ±0.0018 | ±0.0018 | ±0.0022 |
| Fuel Density | ±0.0020 | ±0.0020 | ±0.0017 |
| Eccentricity | negligible | +0.0005 | +0.0032 |
| Total Uncertainty | ±0.0121 | ±0.0039 | ±0.0056 |
| Reference k-infinite | 0.9155 ±0.0121 | 0.9321 ±0.0039 | 0.9436 ±0.0056 |
| Maximum k-infinite | 0.928 | 0.936 | 0.949 |

[1] Calculated for optimum low-density moderation with the 123-group AMPX-KENO code package. The flooded case is the same as Pool A calculated with CASMO-2E.

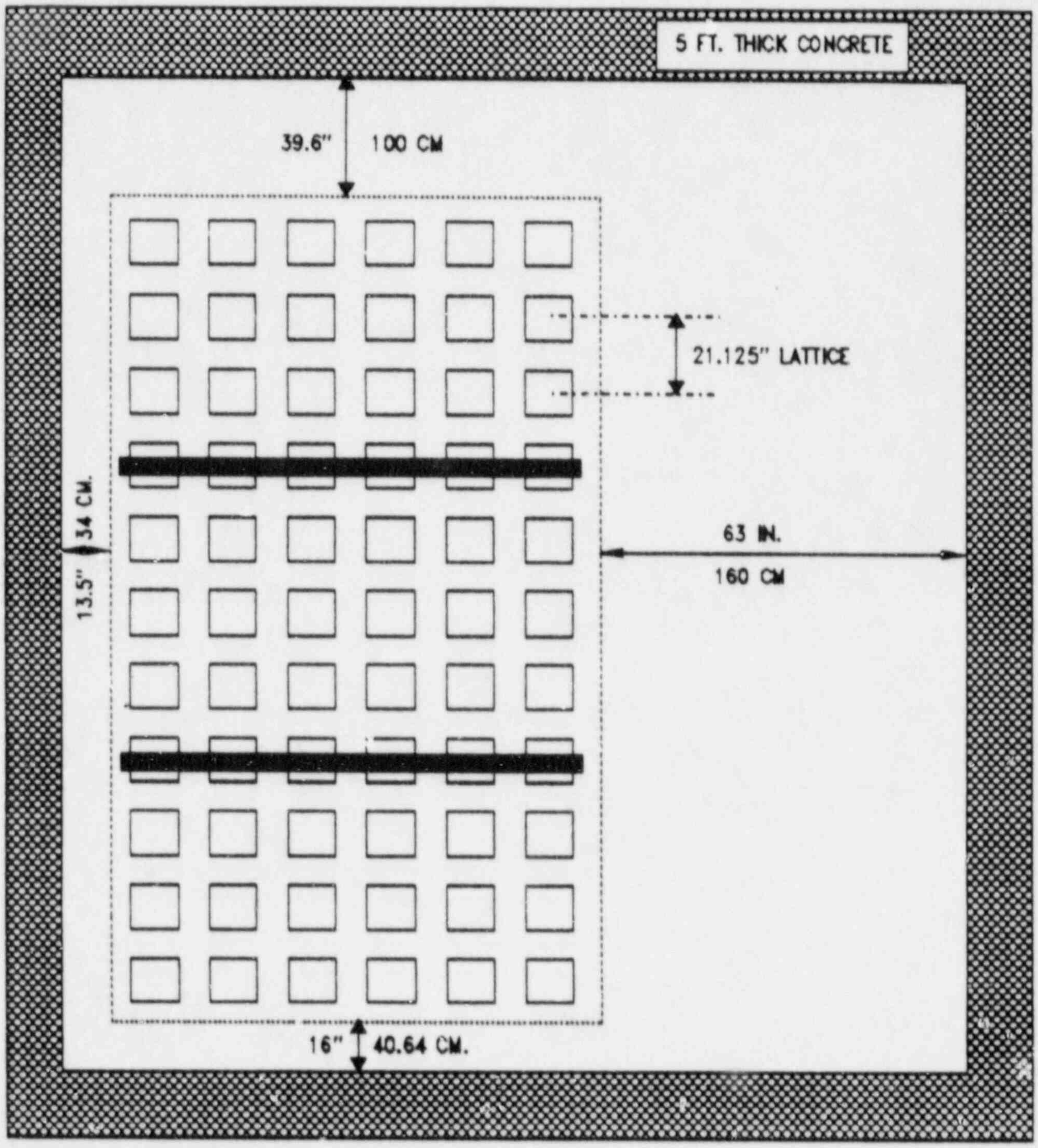


Figure 1  New-Fuel Storage Vault Arrangement

## 3.0 CRITICALITY SAFETY ANALYSES

### 3.1 Reactivity Criteria for Acceptance

For Pool A , Pool B, and the new fuel vault flooded, the criteria for acceptance is that the true k-effective will be equal to or less than 0.95 with a 95% probability at the 95% confidence level, including all known uncertainties, when fully loaded with fuel of the highest anticipated reactivity and flooded with clean unborated water. Under hypothetical accident conditions corresponding to low density optimum moderation, the maximum k-effective of the new fuel storage vault shall not exceed 0.98 including all known uncertainties.

Applicable codes, standards, and regulations include the following:

- General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- NRC letter of April 14,1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications.
- Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Revision 2, December 1981 (Proposed).
- USNRC Standard Review Plan, NUREG-0800, Sections 9.1.1, New Fuel Storage and 9.1.2, Spent Fuel Storage.
- Regulatory Guide 3.41, Validation of Calculational Methods for Nuclear Criticality Safety (and related ANSI Standard N16.9-1975).
- ANSI/ANS 57.2-1983, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.
- ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

## 3.2 ANALYTICAL METHODS

The analytical methods used in the criticality safety analysis are described below. Related benchmark calculations for CASMO-2E and NITAWL-KENO (SCALE cross-section library) are given in Appendix A.

- **AMPX-KENØ**[1], using the 123 group GAM-THERMOS cross-section library and the Nordheim resonance integral treatment in NITAWL for U-238 resonance shielding. This method of analysis has been extensively benchmarked[2] against critical experiments and found to exhibit a bias of 0.000 ± 0.003 (95% probability at a 95% confidence level) plus a small correction for water-gap between assemblies. The 123-group AMPX-KENO was used only for the low moderator density calculations for the new-fuel storage vault where the estimated correction for water-gap results in a bias of 0.0024.

- **Casmo-2E**[3], a multi-group transport theory code for fuel assembly calculations. This method of analysis has been benchmarked (Appendix A) against critical experiments with a bias of 0.0013 ± 0.0018, and was used as the primary calculational method for assemblies stored in the flooded condition.

- **AMPX-KENO,** using the 27-group SCALE cross-section library and the Nordheim routine in NITAWL. (Scale is an acronym for Standardized Computer Analysis for Licensing Evaluations.) This method of analysis was used primarily as a check of the CASMO reference calculations and resulted in lower reactivity values suggesting that CASMO may be over-predicting for the large water gaps of the Pool A and Pool B racks and therefore giving conservative results. Benchmark calculations for the 27-group Scale calculations with AMPX-KENO are given in Appendix A.

- **NULIF**[5]**-PDQ-7**[6], a diffusion theory method of analysis based upon the multi-group cross-section generation code, NULIF, and the two-dimensional diffusion theory code, PDQ-7. This method of analysis was used for a third independent criticality calculation and to investigate certain accident conditions.

## 3.3 Reference Fuel Assembly

The reference design fuel assembly is the Babcock & Wilcox standard 15 x 15 array of $UO_2$ fuel rods with 17 rods replaced by 16 control guide tubes and one central instrument thimble. The principal fuel assembly specifications are listed in Table 2 for the design basis enrichment of 4.3%.

Table 2 FUEL ASSEMBLY DESIGN SPECIFICATIONS

| | |
|---|---|
| **FUEL ROD DATA:** | |
| Outside diameter, in. | 0.430 |
| Cladding inside diameter, in. | 0.377 |
| Cladding material | Zr-4 |
| Cladding density, g/cm$^3$ | 6.588 |
| Pellet diameter, in. | 0.369 ± 0.005 |
| Pellet density, % theoretical | 95% ± 2% |
| Dishing factor | 0.982 |
| Stack density, g $UO_2$/cm$^3$ | 10.225 ± 0.205 |
| Fuel enrichment, wt % U-235 | 4.3 ± .05 |
| **FUEL ASSEMBLY DATA:** | |
| Fuel rod array | 15 x 15 |
| Number of fuel rods | 208 |
| Fuel rod pitch, in. | 0.568 |
| Number of control guide tubes | 16 |
| Outside diameter, in. | 0.530 |
| Inside diameter, in. | 0.498 |
| Number of instrument thimbles | 1 |
| Outside diameter, in. | 0.493 |
| Inside diameter, in. | 0.441 |
| grams U-235 per axial cm | 55.50 |
| Assembly pitch in core, in. | 8.587 |

### 3.4 New Fuel Storage Vault

The new fuel storage vault is a 6 x 11 array of storage locations arranged on a 21 $^1/_4$ inch square lattice spacing as illustrated in Figure 1. Each fuel assembly is supported at the top and bottom and no credit was taken for the small amount of structural material (braces) within the active fuel region of the array. Concrete walls of the storage vault room were assumed to be five-feet thick and located as indicated on the figure. The active fuel begins 22 centimeters above the concrete base and the concrete roof of the vault begins ~130 centimeters above the fuel.

For the hypothetical conditions of low density moderation, criticality analyses were performed in three dimensions with the NITAWL-KENO computer package (Monte Carlo technique) using the 123-group GAM-THERMOS cross-section library: Although no critical experiments are available for the hypothetical low density moderation, Napolitano et al.[4] have compared the 123-group NITAWL-KENO model with continuous energy SAM-CE calculations with good agreement, which provides additional confidence in the calculated k-effective values for the new-fuel storage vault.

Preliminary calculations indicated that, with the new-fuel vault filled with 4.3% enriched fuel in all locations, the calculated reactivity would not provide an adequate subcriticality margin at the postulated low density optimum moderation. By trial and error, it was determined that by leaving 12 storage locations vacant in two symmetric rows of six locations each, the reactivity is reduced sufficiently to provide an adequate safety margin below criticality.

Calculations of the new fuel storage vault under low moderator density conditions were made using the configuration shown in Figure 1 and assuming that 12 storage locations in two rows were vacant. These 12 vacant locations, indicated on Figure 1 as black bars, provide additional leakage that is necessary for the storage vault to satisfy SRP 9.1.1 requirements under low moderator density conditions. Figure 2 shows the calculated k-effective values for various moderator densities and indicates that the optimum moderation (maximum k-effective) occurs at a water density of ~7.5%, with a maximum k-effective of 0.928 for 4.3% enriched fuel (with a one-sided tolerance factor [7] for 95% probability at the 95% confidence level).

Under fully flooded conditions, the new fuel vault is identical in configuration to the Pool A racks and calculations were made with the CASMO-2E program. The maximum k-infinite (infinite array of storage cells) is 0.936 with fuel of 4.3% enrichment.

### 3.5 Pool A Fuel Storage Racks (and Transfer Canal Racks)

The Pool A fuel storage racks are similar to the racks in the new fuel vault, with storage locations arranged on a 21 $^{1}/_{8}$ inch lattice spacing. No credit was taken for the small amount of structural material in the active fuel region of the Pool A racks and criticality control is achieved by water separation alone. The transfer canal storage racks are also on a 21 $^{1}/_{8}$ inch lattice spacing and calculations for Pool A will also apply to the transfer canal racks.

Calculations of the Pool A racks at various temperatures showed that the highest reactivity occurs at room temperature (20°C) and that higher temperatures or the presence of voids reduces the reactivity. At a temperature of 20°C, the nominal reactivity (k-infinite), as calculated by CASMO-2E, is 0.9308 for 4.3% enriched fuel. Check calculations with AMPX-KENO, using 27-group SCALE cross-sections, yielded a bias-corrected k-infinite of 0.9217 $\pm$ 0.0065 (95%/95%), which confirms the CASMO-2E value. With uncertainties added, the maximum reactivity of the Pool A racks of 0.936 for fuel of 4.3% enrichment. The same maximum reactivity value will apply to the transfer canal storage racks.

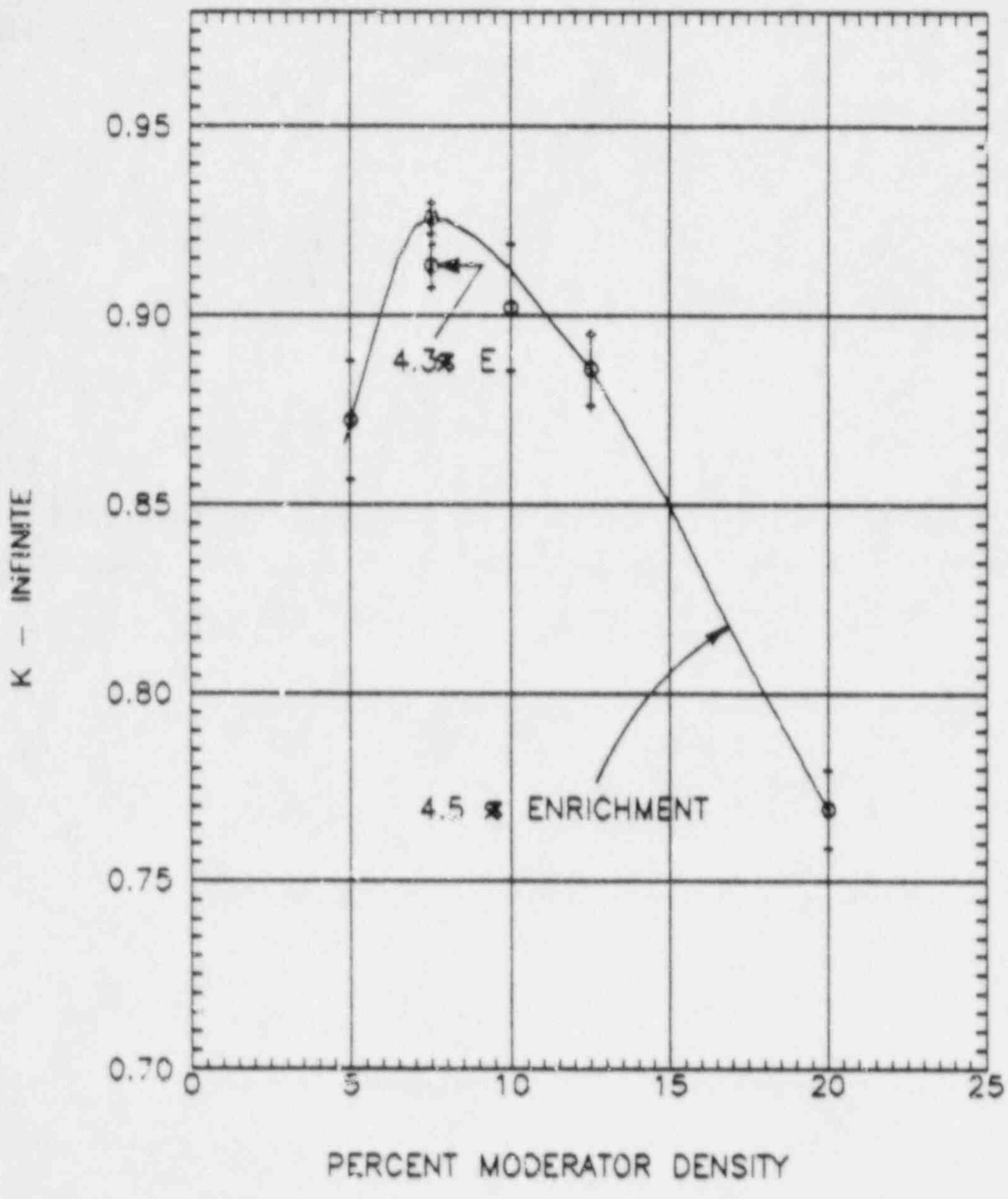


Figure 2 OPTIMUM LOW DENSITY MODERATION

### 3.6 Pool B Fuel Storage Racks

As illustrated in Figure 3, the Pool B storage racks consist of a series of 0.187 inch thick square stainless steel boxes located on a 13.625 inch lattice spacing. The nominal box I.D. is 9.12 inches and the flux-trap water-gap is 4.131 inches.

CASMO-2E calculations showed that reactivity of the Pool B racks increases slightly with temperature, as indicated in the following table. The highest reactivity occurs at 120°C, which is the approximate saturation temperature when boiling will begin at the depth of submergence of the fuel. A temperature of 120°C was therefore used as the reference temperature although it is unlikely that this temperature will ever be reached in practice. Voids introduced by boiling would reduce the reactivity as shown in Table 3.

Table 3 Reactivity effect of Temperature in the Pool B Racks

| Temperature, °C, | Delta-k |
|---|---|
| 20 | -0.0039 |
| 45 | -0.0017 |
| 65 | -0.0009 |
| 90 | -0.0007 |
| 120 | Reference |
| 120 + 10% void | -0.0651 |

Under normal storage conditions, with nominal dimensions, the calculated k-infinite for the Pool B racks with 4.3% enriched fuel is 0.9436 at 120°C. With calculational uncertainties and the reactivity effects of typical manufacturing tolerances added, the maximum reactivity is 0.949 which is within the acceptable limits of the NRC guidelines (k-effective of 0.95). If included, axial leakage in the Pool B racks would slightly reduce this maximum reactivity by -0.003 delta-k.

Independent check calculations with the 27-group SCALE cross-sections in NITAWL-KENO gave a k-infinite of 0.910 ± 0.035 and the NULIF-PDQ7 diffusion theory calculation yielded a k-infinite of 0.940. Both of these independent calculations confirmed the reference CASMO-2E calculation and suggest that CASMO-2E may be conservatively over-predicting reactivity.

### 3.7 Equivalence Factors in Pool B

The maximum reactivity of the Pool B racks is based on the standard B&W fuel assembly with the design specifications shown in Table 2 and includes the effect of conservative manufacturing tolerances. Minor revisions in the design fuel pellet diameter and/or fuel density beyond the tolerance range indicated in Table 2, could, if necessary, be accommodated by adjustment of the allowable enrichment according to reactivity equivalence factors. The equivalence factor is the change in enrichment necessary to compensate for the reactivity effect of a change in either pellet diameter, fuel density or both. An increase in nominal fuel pellet diameter of 1% (e.g. from 0.369 ± 0.005 inches to 0.373 ± 0.005 inches) would require a reduction of 0.035 in percent enrichment (e.g. to 4.265%) to result in approximately the same limiting reactivity. Similarly, an increase in nominal $UO_2$ stack density of 1% (e.g. from 10.225 ± 0.205 g/cc to 10.33 ± 0.205 g/cc), could be compensated by a reduction in allowable enrichment of 0.02% (e.g. to an enrichment of 4.28%).

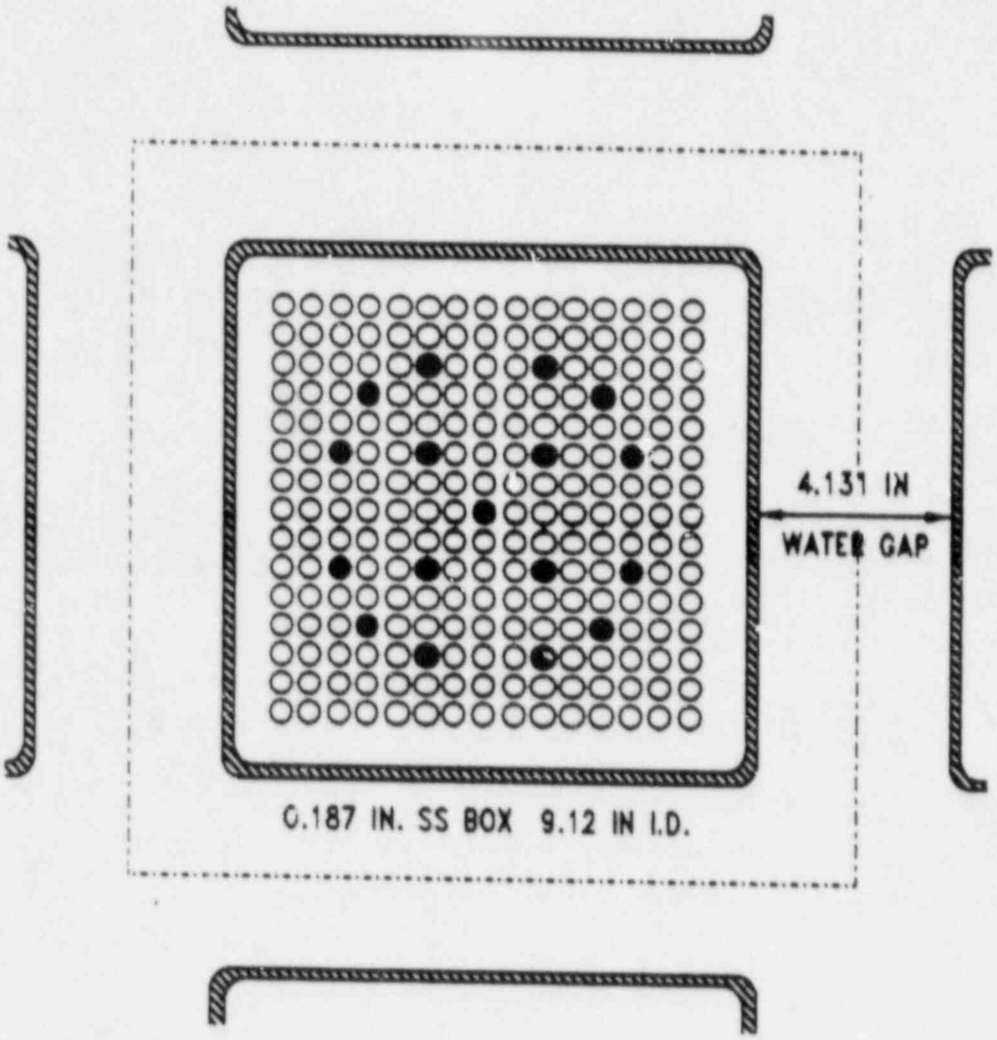


FIGURE 3 CONFIGURATION OF POOL B FUEL STORAGE RACK CELLS

## 4.0 ABNORMAL/ACCIDENT CONDITIONS

### 4.1 NEW FUEL STORAGE VAULT

For the new fuel storage vault, the flooded condition and the low density optimum moderation cases constitute the potential accident conditions that must be considered in the criticality safety evaluation. Under the double contingency principle of ANSI N16.1-1975, invoked by the April 1978 NRC letter, the simultaneous occurrence of other accident conditions need not be considered. In the normally-dry storage conditions of the new fuel vault, the reactivity in the absence of moderator is very low and no additional accident conditions have been identified that would significantly affect the criticality safety margin of the new-fuel storage racks and no additional new or unreviewed safety considerations are introduced by the increase in enrichment.

### 4.2 POOL A RACKS

The effects of credible abnormal or accident conditions on the k-effective of the Pool A storage racks are summarized in Table 4. Of the accident conditions evaluated, only one - the misplacement of a fuel assembly - has the potential for more than a negligible positive reactivity effect. For the accident case of a fresh fuel assembly conservatively assumed to be positioned outside and immediately adjacent to a fuel assembly within the rack, the presence of soluble poison is necessary to preclude exceeding the NRC guideline on reactivity. Two dimensional PDQ-7 calculations of the Pool A rack, using diffusion constants generated by CASMO-2E with various soluble boron concentrations, indicated that a concentration of 600 PPM of soluble boron was adequate to assure a maximum k-infinite less than 0.95 under the postulated accident condition (calculated maximum k-infinite of 0.947). These values also apply to the transfer canal storage racks.

Table 4 Reactivity Effects of Accident Conditions in the Pool A and Transfer Canal Racks

| Condition | Reactivity Effect |
|---|---|
| Temperature Increase | Negative - rack criticality evaluated at temperature of highest reactivity |
| Boiling (Void) | Negative |
| Assembly on Top of Rack | Negligible - separation greater than 12 inches |
| Misplaced Fuel Assembly | Positive - requires minimum of 600 ppm soluble boron |

## 4.3 POOL B RACKS

Potential accident conditions have also been evaluated for the Pool B storage racks and are summarized in Table 5. No conditions have been identified that would result in k-effective exceeding the 0.95 NRC guideline.

Table 5 Reactivity Effects of Accident Conditions in the Pool B Racks

| Condition | Reactivity Effect |
|---|---|
| Temperature Increase | Negligible - rack criticality evaluated at 120°C |
| Boiling (Void) | Negative |
| Assembly on Top of Rack | Negligible - separation greater than 12 inches |
| Misplaced Fuel Assembly | Negligible - there is insufficient space available to insert a misplaced fuel assembly |

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# APPENDIX A

# BENCHMARK CALCULATIONS

## 1. INTRODUCTION AND SUMMARY

The objective of this benchmarking study is to verify both the AMPX (NITAWL)-KENO (Refs. 1 and 2) methodology with the 27-group SCALE cross-section library (Refs. 3 and 4) and the CASMO-2E code (Refs. 5, 6, 7, and 8) for use in criticality calculations of high density spent fuel storage racks. Both calculational methods are based on transport theory and have been benchmarked against critical experiments that simulate typical spent fuel storage rack designs as realistically as possible. Results of these benchmark calculations with both methodologies are consistent with corresponding calculations reported in the literature and with the requirements of Regulatory Guide 3.41,* Rev. 1, May 1977.

Results of these benchmark calculations show that the 27-group (SCALE) AMPX-KENO calculations consistently underpredict the critical eigenvalue by 0.0106 ± 0.0048 $\Delta k$ (with a 95% probability at a 95% confidence level) for critical experiments (Ref. 9) selected to be representative of realistic spent fuel storage rack configurations and poison worths. Similar calculations by Westinghouse (Ref. 11) suggest a bias of 0.012 ± 0.0023, and the results of ORNL analyses of 54 relatively "clean" critical experiments (Ref. 12) show a bias of 0.0100 ± 0.0013.

Similar calculations with CASMO-2E for clean critical experiments resulted in a bias of 0.0013 ± 0.0018 (95%/95%). CASMO-2E and AMPX-KENO intercomparison calculations of infinite arrays of poisoned cell configurations show very good agreement and suggest that a bias of 0.0013 ± 0.0018 is the reasonably expected bias and uncertainty for CASMO-2E calculations.

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*Validation of Calculational Methods for Nuclear Criticality Safety. (See also ANSI N16.9-1975.)

The benchmark calculations reported here indicate that either the 27-group (SCALE) AMPX-KENO or CASMO-2E calculations are acceptable for criticality analysis of high density spent fuel storage racks. The preferred methodology, however, is to perform independent calculations with both code packages and to utilize the higher, more conservative value for the reference design infinite multiplication factor.

## 2. AMPX (NITAWL)-KENO BENCHMARK CALCULATIONS

Analysis of a series of Babcock & Wilcox (B&W) critical experiments (Ref. 9), which include some with absorber sheets typical of a poisoned spent fuel rack, is summarized in Table 1, as calculated with AMPX-KENO using the 27-group SCALE cross-section library and the Nordheim resonance integral treatment in NITAWL. The mean for these calculations is 0.9894 ± 0.0019, conservatively assuming the larger standard deviation calculated from the $k_{eff}$ values. With a one-sided tolerance factor corresponding to 95% probability at a 95% confidence level (Ref. 10), the calculational bias is +0.0106 with an uncertainty of ±0.0048.

Similar calculational deviations reported by Westinghouse (Ref. 11) are also shown in Table 1 and suggest a bias of 0.012 ± 0.0023 (95%/95%). In addition, ORNL (Ref. 12) has analyzed some 54 critical experiments using the same methodology, obtaining a mean bias of 0.0100 ± 0.0013 (95%/95%). These published results are in good agreement with the results obtained in the present analysis and lend further credence to the validity of the 27-group AMPX-KENO calculational model for use in criticality analysis of high density spent fuel storage racks. Variance analysis of the data in Table 1 suggests the possibility that an unknown factor may be causing a slightly larger variance than might be expected from the Monte Carlo statistics alone. However, such a

Table 1

RESULTS OF 27-GROUP (SCALE) AMPX-KENO CALCULATIONS
OF B&W CRITICAL EXPERIMENTS

| Experiment Number | Calculated $k_{eff}$ | $\sigma$ | Westinghouse Calculated-meas. $k_{eff}$ |
|---|---|---|---|
| I | 0.9889 | ±0.0049 | -0.008 |
| II | 1.0040 | ±0.0037 | -0.012 |
| III | 0.9985 | ±0.0046 | -0.008 |
| IX[(1)] | 0.9924 | ±0.0046 | -0.016 |
| X | 0.9907 | ±0.0039 | -0.008 |
| XI | 0.9989 | ±0.0044 | +0.002 |
| XII | 0.9932 | ±0.0046 | -0.013 |
| XIII | 0.9890 | ±0.0054 | -0.007 |
| XIV | 0.9830 | ±0.0038 | -0.013 |
| XV | 0.9852 | ±0.0044 | -0.016 |
| XVI | 0.9875 | ±0.0042 | -0.015 |
| XVII | 0.9811 | ±0.0041 | -0.015 |
| XVIII | 0.9784 | ±0.0050 | -0.015 |
| XIX | 0.9888 | ±0.0033 | -0.016 |
| XX | 0.9922 | ±0.0048 | -0.011 |
| XXI | 0.9783 | ±0.0039 | -0.017 |
| Mean | 0.9894 | ±0.0011[(2)] | -0.0120 ± 0.0010 |
| Bias | 0.0106 | ±0.0019[(3)] | 0.0120 ± 0.0010 |
| Bias (95%/95%) | 0.0106 | ±0.0048 | 0.0120 ± 0.0023 |
| Maximum Bias | 0.0154 | | 0.0143 |

(1) Experiments IV through VIII used $B_4C$ pin absorbers and were not considered representative of poisoned storage racks.
(2) Calculated from individual standard deviations.
(3) Calculated from $k_{eff}$ values and used as reference.

factor, if one truly exists, is too small to be resolved on the basis of critical-experiment data presently available. No trends in $k_{eff}$ with intra-assembly water gap, with absorber sheet reactivity worth, or with soluble poison concentration were identified.*

## 3. CASMO-2E BENCHMARK CALCULATIONS

### 3.1 GENERAL

The CASMO-2E code is a multigroup transport theory code utilizing transmission probabilities to accomplish two-dimensional calculations of reactivity and depletion for BWR and PWR fuel assemblies. As such, CASMO-2E is well-suited to the criticality analysis of spent fuel storage racks, since general practice is to treat the racks as an infinite medium of storage cells, neglecting leakage effects.

CASMO-2E is closely analogous to the EPRI-CPM code (Ref. 13) and has been extensively benchmarked against hot and cold critical experiments by Studsvik Energiteknik (Refs. 5, 6, 7, and 8). Reported analyses of 26 critical experiments indicate a mean $k_{eff}$ of 1.000 ± 0.0037 (1σ). Yankee Atomic (Ref. 14) has also reported results of extensive benchmark calculations with CASMO-2E. Their analysis of 54 Strawbridge and Barry critical experiments (Ref. 15) using the reported buckling indicates a mean of 0.9987 ± 0.0009 (1σ), or a bias of 0.0013 ± 0.0018 (with 95% probability at a 95% confidence level). Calculations were repeated for seven of the Strawbridge and Barry experiments

*Significantly large trends in $k_{eff}$ with water gap and with absorber sheet reactivity worth have been reported (Ref. 16) for AMPX-KENO calculations with the 123-group GAM-THERMOS library.

selected at random, yielding a mean $k_{eff}$ of 0.9987 ± 0.0021 (1σ), thereby confirming that the cross-section library and analytical methodology being used for the present calculations are the same as those used in the Yankee analyses. Thus, the expected bias for CASMO-2E in the analysis of "clean" critical experiments is 0.0013 ± 0.0018 (95%/95%).

## 3.2 BENCHMARK CALCULATIONS

CASMO-2E benchmark calculations have also been made for the B&W series of critical experiments with absorber sheets, simulating high density spent fuel storage racks. However, CASMO-2E, as an assembly code, cannot directly represent an entire core configuration* without introducing uncertainty due to reflector constants and the appropriateness of their spectral weighting. For this reason, the poisoned cell configurations of the central assembly, as calculated by CASMO-2E, were benchmarked against corresponding calculations with the 27-group (SCALE) AMPX-KENO code package. Results of this comparison are shown in Table 2. Since the differences are well within the normal KENO statistical variation, these calculations confirm the validity of CASMO-2E calculations for the typical high density poisoned spent fuel rack configurations. The differences shown in Table 2 are also consistent with a bias of 0.0013 ± 0.0018, determined in Section 3.1 as the expected bias and uncertainty of CASMO-2E calculations.

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*Yankee has attempted such calculations (Ref. 14) using CASMO-2E-generated constants in a two-dimensional, four-group PDQ model, obtaining a mean $k_{eff}$ of 1.005 for 11 poisoned cases and 1.009 for 5 unpoisoned cases. Thus, Yankee benchmark calculations suggest that CASMO-2E tends to slightly overpredict reactivity.

Table 2

RESULTS OF CASMO-2E BENCHMARK (INTERCOMPARISON) CALCULATIONS

| B&W Experiment No.[1] | $k_\infty$[1] AMPX-KENO[2] | CASMO-2E | $\Delta k$ |
|---|---|---|---|
| XIX | 1.1203 ± 0.0032 | 1.1193 | 0.0010 |
| XVII | 1.1149 ± 0.0039 | 1.1129 | 0.0020 |
| XV | 1.1059 ± 0.0038 | 1.1052 | 0.0007 |
| Interpolated[3] | 1.1024 ± 0.0042 | 1.1011 | 0.0013 |
| XIV | 1.0983 ± 0.0041 | 1.0979 | 0.0004 |
| XIII | 1.0992 ± 0.0034 | 1.0979 | 0.0013 |
| Mean<br>Uncertainty | ± 0.0038 | | 0.0011<br>±0.0006 |
| BWR fuel rack | 0.9212 ± 0.0027 | 0.9218 | -0.006 |

(1) Infinite array of central assemblies of 9-assembly B&W critical configuration (Ref. 9).
(2) $k_\infty$ from AMPX-KENO corrected for bias of 0.0106 $\Delta k$.
(3) Interpolated from Fig. 28 of Ref. 9 for soluble boron concentration at critical condition.

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16. S. E. Turner and M. K. Gurley, "Evaluation of AMPX-KENO Benchmark Calculations for High Density Spent Fuel Storage Racks," Nuclear Science and Engineering, 80(2): 230-237, February 1982.

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

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Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 180

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This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY: H D Hukill
Vice President & Director, TMI-1

Sworn and Subscribed
to before me this 12th
day of January, 1988.

Sharon P. Brown
Notary Public

SHARON P. BROWN, NOTARY PUBLIC
MIDDLETOWN BORO, DAUPHIN COUNTY
MY COMMISSION EXPIRES JUNE 12, 1989
Member, Pennsylvania Association of Notaries

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I. Technical Specification Change Request (TSCR) No. 180

GPUN requests that the following changed replacement pages be inserted into the existing Technical Specifications:

Revised pages: 4-1, 4-2, 4-10, 5-6 and 5-7.

These replacement pages are attached to this TSCR.

II. Reason For Change

The change in maximum allowable fuel enrichment for new fuel storage at TMI-1 being proposed herein is in support of cycle 7 operation and subsequent cycles of operation which currently plan to use fuel loadings of higher enrichment. These fuel loadings of higher enrichment would allow for longer operational cycle lengths.

III. Safety Evaluation Justifying The Change

The proposed Technical Specifications incorporate appropriate surveillance and design requirements to allow for the storage of fuel with an enrichment not to exceed 4.3 w/o U-235 in the TMI-1 New Fuel Storage Vault, Fuel Transfer Canal, Spent Fuel Pool "A" and Spent Fuel Pool "B". The attached criticality safety analysis verifies that the higher enriched fuel can be stored in these locations without exceeding the NRC guidelines on $K_{effective}$ under normal and accident conditions. To ensure that the NRC guidelines on $K_{effective}$ are met at all times, two (2) special restrictions are required. These restrictions appear below:

1. The restriction to leave twelve (12) storage locations in the New Fuel Storage Vault vacant (aligned in two rows of six locations each; transverse rows numbers four and eight) of fissile or moderating material. The restriction will ensure that the NRC Standard Review Plan (NUREG 0800) Section 9.1.1 requirements for reactivity under hypothetical conditions of low density "optimum" moderation are met by allowing for the necessary additional neutron leakage.

2. The restriction to maintain at least 600 ppm soluble boron in the Spent Fuel Pool "A" and the Fuel Transfer Canal during new fuel movements in or over the pool or canal when new fuel is being stored in the pool or canal. This will ensure that the maximum reactivity is less than the NRC maximum allowed reactivity value for the postulated accident condition of a misplaced fuel assembly located outside the rack but immediately adjacent to a fuel assembly within the rack.

Technical Specification Section 5.4.1(a) is being revised to indicate that 4.3 w/o U-235 new fuel can be stored in the new fuel storage vault or spent fuel pools without exceeding a $K_{effective}$ of .95. The currently existing Section 5.4.1(a) requires that a $K_{effective}$ of less than .9 be maintained. The .9 $K_{effective}$ criteria was the NRC pre-1978 limit based on the fact that uncertainties were not considered in the criticality analyses. However, the current NRC guidelines on $K_{effective}$ for new fuel storage (NRC Standard Review Plan 9.1.1) require the consideration of uncertainties in criticality analyses and therefore the required $K_{effective}$ is increased appropriately. The proposed Technical Specification Section 5.4.1(a) recognizes the revised criteria.

Technical Specification Section 5.4.1(a) is being revised to indicate the two (2) restrictions concerning new fuel storage. In addition to the revision to 5.4.1(a), the two (2) restrictions will be included in the appropriate plant procedures.

Technical Specification Section 5.4.1(a) also is being revised to identify the proper fuel rack nominal center-to-center spacings for the Spent Fuel Pool "B" racks.

Technical Specification Section 5.4.1(b) is being revised to indicate a restriction concerning new fuel manipulation in the fuel transfer canal when new fuel is being stored there. In addition to the revision to 5.4.1(b), the restriction will be included in the appropriate plant procedures.

Technical Specification Section 5.4.2(d) is being revised to add a note indicating that, of the 66 storage locations in the new fuel vault racks, twelve (12) of the locations are required to be vacant of fissile or moderating material.

Technical Specification Section 5.4.2(f) is being revised to specify the maximum allowable grams of U-235 per axial centimeter of fuel assembly. This change is necessary to support the increase to 4.3 w/o U-235 new fuel.

Technical Specification Section 4.1 Bases is being revised to include a discussion concerning a minimum boron concentration for the Spent Fuel Pool.

Technical Specification Table 4.1-3 is being revised to check that the boron concentration is greater than or equal to 660 ppmb.

## IV. No Significant Hazards Considerations

GPUN has determined that the Technical Specification Change Request poses no significant hazards as defined by the NRC in 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. There are no design basis events in TMI-1 FSAR Chapter 14 or elsewhere which are affected by this proposed amendment. Also, an analysis has been performed and has demonstrated that the NRC criticality requirements for the storage of new fuel have been met under both normal and abnormal conditions.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The only event of concern with respect to storage of new fuel is criticality and as mentioned in item (1) above, an analysis has demonstrated that the proposed amendment would not result in any kind of criticality event.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The safety criteria contained in the Technical Specification Bases are not impacted by this proposed amendment.

The Commission has provided guidelines pertaining to the application of the three (3) standards by listing specific examples in 48 FR 14870. The proposed amendment is considered to be in the same category as example (vi) of amendments that are considered not likely to involve significant hazards considerations in that the result of this proposed amendment is clearly within all acceptance criteria with respect to the Standard Review Plan.

## V. Implementation

It is requested that the amendment authorizing this change become effective no later than May 1, 1988. This is needed to support the receipt of new fuel at TMI-1 for cycle 7 operation. Delay beyond this date could adversely impact the scheduled TMI-1 refueling outage and the shipment of damaged TMI-2 fuel offsite.

## VI. Amendment Fee (10 CFR 170.21)

Pursuant to the provisions of 10 CFR 170.21, attached is a check for $150.00.

## 4. SURVEILLANCE STANDARDS

During Reactor Operational Conditions for which a Limiting Condition for Operation does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/component operable, the associated surveillance requirement must be current. The above applicability requirements assure the operability of systems/components for all Reactor Operating Conditions when required by the Limiting Conditions for Operation.

## 4.1 OPERATIONAL SAFETY REVIEW

### Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

### Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

### Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system, engineered safety feature protection system, and heat sink protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Each post accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies shown in Table 4.1-4.

### Bases

#### Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.



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Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required monthly on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protection channels is as follows:

| | |
|---|---|
| Channels A, B, C & D | Before Startup, when shutdown greater than 24 hours |
| Channel A | One Week After Startup |
| Channel B | Two Weeks After Startup |
| Channel C | Three Weeks After Startup |
| Channel D | Four Weeks After Startup |

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested monthly. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protection channels and the regulating control rod power SCRs shall be trip tested prior to startup when the reactor has been shutdown for greater than 24 hours.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

TABLE 4.1-3 Cont'd.

| | Item | Check | Frequency |
|---|---|---|---|
| 4. | Spent Fuel Pool Water Sample | Boron concentration greater than or equal to 600 ppmb | Monthly and after each makeup. |
| 5. | Secondary Coolant System Activity | Isotopic analysis for DOSE EQUIVALENT I-131 concentration | At least once per 72 hours when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F |
| 6. | Boric Acid Mix Tank or Reclaimed Boric Acid Tank | Boron concentration | Twice weekly*** |
| 7. | Deleted | | |
| 8. | Deleted | | |
| 9. | Deleted | | |
| 10. | Sodium Hydroxide Tank | Concentration | Quarterly and after each makeup. |
| 11. | Deleted | | |
| 12. | Deleted | | |

# Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** Deleted

*** The surveillance of either the Boric Acid Mix Tank or the Reclaimed Boric Acid Tank is not necessary when that respective tank is empty.

## 5.4 NEW AND SPENT FUEL STORAGE FACILITIES

### Applicability

Applies to storage facilities for new and spent fuel assemblies.

### Objective

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

### Specification

5.4.1 NEW FUEL STORAGE

a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions for the new fuel storage vault and the Spent Fuel Pool "A". The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 13-5/8 inches in both directions for the Spent Fuel Pool "B". This spacing is sufficient to maintain a K effective of less than .95 based on fuel assemblies with an enrichment of 4.3 weight percent $U^{235}$. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition. When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to ensure meeting the NRC maximum allowable reactivity value under the postulated accident condition of a misplaced fuel assembly.

b. New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in an 8 x 8 array storage rack having a nominal center to center distance of 21-1/8 inches. When fuel is being moved in or over the fuel transfer canal, a boron concentration of at least 600 ppmb must be maintained to ensure that, under the postulated accident condition of a misplaced fuel assembly, the maximum reactivity will be less than the NRC maximum allowable reactivity. This applies only when fuel is being stored in the canal.

c. New fuel may also be stored in shipping containers.

### 5.4.2 SPENT FUEL STORAGE

a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.

b. Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.

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

d. The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:

| | South End of Fuel Transfer Canal RB | Spent Fuel Pool A North End of Fuel Handling Building | Spent Fuel Pool B South End of Fuel Handling Building | Dry New Fuel Storage Area Fuel Handling Building |
|---|---|---|---|---|
| Fuel Assys | 64 * | 256 ** | 496 *** | 66**** |
| Cores | 0.36 | 1.45 | 2.8 | 0.37 |

NOTES:
* Includes one space for accommodating a failed fuel detection container.
** Includes three spaces for accommodating failed fuel containers.
*** Spent Fuel Pool B contains spent fuel storage racks with a reduced center-to-center spacing of 13 5/8 inches to increase the storage capacity of the pool.
**** Includes twelve spaces which are required to be vacant of fissile or moderating material so that there is sufficient neutron leakage.

e. All of the fuel assembly storage racks provided are designed to Seismic Class 1 criteria to the accelerations indicated below:

| | Fuel Transfer Canal in Reactor Building | Fuel Handling Building Dry New Fuel Storage Area And Spent Fuel Pool A | Fuel Handling Building Spent Fuel Pool B |
|---|---|---|---|
| Horiz. | 0.76 g | 0.38 g | * |
| Vertical | 0.51 g | 0.25 g | * |

* The "B" pool fuel storage racks are designed using the floor response spectra of the Fuel Handling Building.

f. Fuel in the storage pool shall have a U-235 loading equal to or less than 57.8 grams of U-235 per axial centimeter of fuel assembly.

REFERENCES

(1) FSAR, Section 9.7