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U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

**DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT**
TECHNICAL SPECIFICATION CHANGE REQUEST
SPENT FUEL POOL BORON CONCENTRATION

The enclosure to this letter proposes Technical Specifications changes that increase the limits on stored fuel enrichment, impose a spent fuel boron concentration requirement whenever fuel is stored in the spent fuel pool, and require that the spent fuel pool boron concentration be verified weekly. The increased enrichment limit is necessary to support flexibility in reactor core design. Crediting soluble boron in the spent fuel pool results in more flexible fuel storage constraints for the Palisades fuel storage racks.

Enclosure 1 contains the discussions associated with the proposed changes, the proposed pages, and the existing pages marked to show the proposed changes. Enclosure 2 contains the engineering analyses which form the technical basis for the proposed changes.

Consumers Energy requests this Technical Specification change be approved on or before October 1, 2001; and that 90 days be granted after approval for implementation to accomplish the necessary procedure changes.

A copy of this letter has been sent to the appropriate official of the State of Michigan.

A001

SUMMARY OF COMMITMENTS

This letter completes our commitment, made in our September 29, 2000 letter, to pursue changes to the Technical Specifications which will eliminate the need to credit Boraflex Poison in Palisades Region II spent fuel pool racks. The proposed Technical Specifications changes will eliminate all reliance on the use of Boraflex for reactivity control in the spent fuel pool. Approval of this submittal will close our commitment, made in our December 8, 1997 letter and re-iterated in our September 29, 2000 letter, to perform blackness testing of spent fuel racks if silica concentration in the spent fuel pool indicates significant Boraflex degradation.

Daniel J. Malone
Director, Engineering

CC: Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades
Lou Brandon, Michigan Department of Environmental Quality

Enclosures

CONSUMERS ENERGY COMPANY

TECHNICAL SPECIFICATION CHANGE REQUEST
**SPENT FUEL POOL BORON CONCENTRATION**

To the best of my knowledge, the content of this Technical Specifications change request, which proposes changes to the Technical Specifications which implement Spent Fuel Pool boron concentration requirements, is truthful and complete.

Daniel J. Malone
Director, Engineering

Sworn and subscribed to before me this 2nd day of March 2001

Norma Jean Fowler

Norma Jean Fowler, Notary Public
Van Buren County, Michigan
My commission expires May 14, 2002

(Seal)

# ENCLOSURE 1

**CONSUMERS ENERGY COMPANY**
**PALISADES PLANT**
**DOCKET 50-255**

**TECHNICAL SPECIFICATION CHANGE REQUEST**

**SPENT FUEL POOL BORON CONCENTRATION**

40 Pages

**CONSUMERS ENERGY COMPANY**
**DOCKET 50-255**
**LICENSE DPR-20**

**ENCLOSURE 1**
**REQUEST FOR CHANGE TO THE TECHNICAL SPECIFICATIONS**
**SPENT FUEL POOL BORON CONCENTRATION**

It is requested that the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below.

Attachment 1 to this change request contains the proposed Technical Specifications pages. Attachment 2 contains existing pages marked to show the proposed change. Deleted text is shown as strike-out; added text is shown with a shaded background. Engineering Analysis EA-SFP-99-03, the technical basis for the changes proposed, is contained in Enclosure 2.

As used within this change request, unless otherwise stated, "enrichment" means "maximum planar average U-235 enrichment" and "Region I or Region II refers to Region I or Region II of the fuel storage racks in the Palisades Spent Fuel Pool.

I. Introduction

The purpose of this license amendment request is to incorporate into the Palisades Technical Specifications more flexible fuel loading constraints for the Palisades new fuel storage racks, Region I fuel storage racks and Region II fuel storage racks. Allowed uranium enrichments for storage are increased. The proposed revisions rely on criticality analyses which credit soluble boron for the control of reactivity in the spent fuel pool. The credit for soluble boron in the criticality analyses combined with increased burnup credit allow the neutron absorbing Boraflex material in the Region II fuel storage racks to be ignored. The proposed Technical Specification changes ensure that the presence of the required boron concentration is maintained, therefore ensuring at a 0.95 probability and a 95% confidence level (95/95) that the Palisades fuel pool $k_{eff}$ remains less than 0.95.

The criticality analyses for the Palisades new fuel storage racks, the Region I and Region II fuel storage racks, and the fuel inspection and transfer machinery is documented in EA-SFP-99-03, "*Palisades New Fuel Storage, Fuel Pool and Fuel Handling Criticality Safety Analysis*". The Palisades criticality analyses closely follow the approach described in Westinghouse Spent Fuel Rack Criticality Analysis Methodology (WCAP-14416-NP-A), however the Monte Carlo code package, MONK, is used instead of the KENO code package used by Westinghouse. EA-SFP-99-03 concludes that the 95/95 $k_{eff}$ remains below 0.95 under all normal storage and handling scenarios as well as all credible accident conditions. The analyses take credit for a fuel pool boron concentration of 850 ppm under normal conditions and 1350 ppm under accident conditions. The proposed Technical Specifications maintain the requirement that the fuel pool boron concentration be maintained at or above 1720 ppm whenever fuel is stored in the fuel pool.

On October 28, 1997, pursuant to 10 CFR 70.24(d)(1), the Nuclear Regulatory Commission granted Palisades an exemption from certain requirements for criticality monitors. The basis for the exemption was that inadvertent criticality was not a credible event, and that Palisades new and spent fuel storage facilities met the requirements of 10 CFR 50.68. As shown in the enclosed analyses, EA-SFP-99-03, the Palisades new and spent fuel storage facilities, with credit for soluble boron, provide storage that will continue to meet 10 CFR 50.68 requirements. Therefore, Consumer Energy considers the 1997 exemption from 10 CFR 70.24 requirements will remain valid after approval of the Technical Specifications changes requested herein.

The Technical Specification changes supported by EA-SFP-99-03 fall in three general areas:

A. Allow storage of un-irradiated fuel up to 4.95 wt% enrichment in the new fuel storage racks assuming defined loading patterns.

B. Allow storage of un-irradiated or irradiated fuel up to 4.95 wt% enrichment in Region I fuel storage racks with no credit for soluble boron in the pool under normal conditions, and credit for 1350 ppm of soluble boron under accident conditions.

C. Allow storage of un-irradiated fuel up to 1.14 wt% enrichment and irradiated fuel of equivalent reactivity up to 4.6 wt% initial enrichment in Region II fuel storage racks with credit for 850 ppm of soluble boron in the pool under normal conditions, and credit for 1350 ppm of soluble boron under accident conditions. Assembly burnup and subsequent decay time are also considered in the criticality calculations. The Region II fuel storage rack criticality analysis conservatively ignores the Boraflex poison material present in the racks.

II. Summary of Safety Analyses EA-SFP-99-03

Engineering Analysis EA-SFP-99-03 contains three major areas of criticality analyses associated with this Technical Specifications change request: the new fuel storage racks, Region I fuel storage racks, and Region II fuel storage racks.

A. New Fuel Storage Racks

EA-SFP-99-03 documents the criticality analysis which shows with a 0.95 probability at a 95% confidence level (95/95) that the new fuel storage array $k_{eff}$ remains below 0.95 assuming the rack is fully loaded with 36 un-irradiated assemblies with enrichment $\leq$ 4.05 wt% U-235. EA-SFP-99-03 also shows that the new fuel storage array 95/95 $k_{eff}$ remains below 0.95, assuming the rack is only partially loaded with 24 un-irradiated assemblies with enrichment $\leq$ 4.95 wt% U-235. The center row of the rack is kept empty in this situation. For both configurations, the 95/95 $k_{eff}$ is below 0.95 including all uncertainties and applicable biases for all normal and credible abnormal storage conditions.

B. Region I Fuel Storage Racks

EA-SFP-99-03 documents the criticality analysis which shows that for the Region I storage array the 95/95 $k_{eff}$ remains below 0.95, assuming the rack is fully loaded with un-irradiated or irradiated fuel assemblies with enrichment $\leq$ 4.95 wt% U-235. The 95/95 $k_{eff}$ is shown below 0.95 including all uncertainties and applicable biases for all normal and credible abnormal storage conditions and 0.0 ppm boron in the pool water. Credit is taken for 1350 ppm of boron in the spent fuel pool water when analyzing accident conditions (Double Contingency Principle of ANS/ANSI 8.1).

C. Region II Fuel Storage Racks

EA-SFP-99-03 documents the criticality analysis which shows that the Region II storage array 95/95 $k_{eff}$ remains below 0.95 assuming:

1. Either:

   a. The rack is fully loaded with unirradiated fuel assemblies with enrichment ≤ 1.14 wt% U-235, or

   b. Irradiated fuel assemblies with initial enrichments ≤ 4.60 wt% U-235 and burnup and decay time combinations ensuring $k_{eff}$ < 1.0 assuming 0.0 ppm boron.

2. Credit for 850 ppm of boron in the pool water to ensure the 95/95 $k_{eff}$ is below 0.95 under all normal storage conditions including all uncertainties and applicable biases.

Credit is taken for an additional 500 ppm of boron (1350 ppm total) to show the 95/95 $k_{eff}$ is less than 0.95 under all credible accident conditions (Double Contingency Principle of ANS/ANSI 8.1).

In addition, EA-SFP-99-03 also documents the criticality analyses for handling and storage of un-irradiated or irradiated fuel up to 4.95 wt% enrichment in the Palisades fuel elevator and fuel transfer machine. No Technical Specifications concerning the elevators or transfer machine equipment exist or are proposed. Their design is documented in Section 9.11 of the Palisades Final Safety Analysis Report. Changes to the design calculations allowing handling of fuel up to 4.95 wt% enrichment are evaluated under 10 CFR 50.59. The evaluation has determined that no unreviewed safety questions concerning the elevator or transfer machine criticality analysis exist and, therefore, that NRC review is not required for increasing the allowed enrichment level in this equipment. Fuel elevator and transfer machine criticality analyses are included in EA-SFP-99-03 to provide complete documentation of the Palisades design basis criticality calculations.

The criticality analyses which are the basis for this submittal take credit for the soluble boron in the spent fuel pool water to control the subcritical condition of the irradiated fuel array. The NRC has documented the requirements for use of soluble boron in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998. The precedent of taking credit for soluble boron in spent fuel pool water to provide criticality control has already been established. Soluble boron credit was used in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-14416-NP-A and that methodology was approved for use by an NRC Safety Evaluation dated October 25, 1996.

The utilization of soluble boron, which is normally contained in the spent fuel pool, provides a simple, direct method of ensuring subcriticality. This control feature retains the necessary criticality requirements and has many benefits. Palisades currently takes credit for soluble boron to maintain subcriticality during Fuel Transfer Machine usage and loading of Spent Fuel Storage Casks, and to compensate for increases in reactivity due to fuel handling accidents (double contingency principle of ANS/ANSI 8.1)

# SPENT FUEL POOL BORON CONCENTRATION



While the proposed license amendment assumes the use of soluble boron in the spent fuel pool criticality analysis, the fuel storage configuration specified ensures that the calculated $k_{eff}$ will be less than 1.0 including uncertainties due to manufacturing tolerances and assuming 0.0 ppm of soluble boron in the spent fuel pool. Credit for soluble boron is used to offset uncertainties related to reactivity equivalencing and to increases in reactivity during accident scenarios. Crediting soluble boron provides subcritical margin such that the spent fuel pool $k_{eff}$ is maintained less than or equal to 0.95.

Possible boron dilution events were analyzed to demonstrate that sufficient time is available to detect and mitigate any dilution of the spent fuel pool before the 0.95 $k_{eff}$ design basis is exceeded. The boron dilution evaluation included consideration of the following plant specific features:

1. Spent Fuel Pool and Related System Features
   - Dilution Sources
   - Dilution Flow Rates
   - Boration Sources
   - Instrumentation
   - Administrative Procedures
   - Piping

2. Boron Dilution Initiating Events

3. Boron Dilution Times and Volumes

The results of the spent fuel pool boron dilution evaluation are summarized in EA-SFP-99-03. As part of the evaluation, available dilution sources were compiled and evaluated against the calculated dilution volumes to determine the potential of a spent fuel pool boron dilution event. For each dilution scenario, calculations were performed to define the dilution time for the spent fuel pool to reach 850 ppm. The evaluation shows that a large volume of water (123,007 gallons) is necessary to dilute the spent fuel pool from the present Technical Specification Limit of 1720 ppm to a soluble boron concentration where a $k_{eff}$ of 0.95 would be approached in the pool. For the limiting dilution source flow rate the dilution time to reach a pool concentration of 850 ppm was determined to be 9.8 hours.

The first 15,000 gallons of dilution water would fill the pool to its overflow level. The remaining 107,600 gallons needed dilute the pool to 850 ppm would all be over boarded onto the pool deck and down the equipment hatch, elevator shaft, or the stair well, all of which are located within 4 to 10 feet of the pool. The resulting water distribution throughout the auxiliary building and safeguards room basement would result in high sump level alarms in the control room. The large amounts of water on the floor would be easily spotted by the operators whether they have specifically been sent there in response to an alarm or if they were making normal rounds through the aux building and fuel pool on a shiftly basis. Therefore, it is reasonable to assume that the operators will recognize and terminate this event well before the boron concentration in the spent fuel pool drops below 850 ppm at 9.8 hours into the event.

The evaluation shows that the dilution of the spent fuel pool boron concentration will be terminated before it reaches the 850 ppm limit. The dilution calculation combined with the criticality calculation, which shows that the spent fuel rack $k_{eff}$ will remain below 1.0 even when flooded with unborated water (0 ppm), provide a level of safety comparable to the conservative criticality analysis methodology used in prior fuel storage criticality calculations.

III. Proposed Changes

This License Amendment Request proposes revisions to the Technical Specifications associated with controlling the storage of assemblies with higher initial enrichments, different enrichment and burnup combinations, and the consideration of decay time. The proposed Technical Specification changes also include changes to some Limiting Conditions for Operation, Surveillance Requirements, and plant procedure changes that enhance the control of the boron concentration in the spent fuel pool under normal and accident conditions. The following sections detail the proposed changes and provide a short explanation of the purpose of the change.

A. LCO 3.7.15, Spent Fuel Pool (SFP) Boron Concentration.

1. Change the Applicability from:

   *"When fuel assemblies are stored in the SFP and a verification of the stored assemblies has not been performed."*

   to:

   *"When fuel assemblies are stored in the Spent Fuel Pool."*

   This is a more restrictive change.

2. Delete Required Action A.2.2 since verification alone would no longer restore the plant to analyzed conditions. Required Action A.2.1 is renumbered to "A.2."

The existing LCO is aimed at protecting against criticality during a fuel handling accident or misloading event. Criticality analyses which are the basis for this license amendment request credit boron for normal storage as well as for accident scenarios. Therefore, the applicability of Section 3.7.15 is extended to all times when fuel assemblies are stored in the Palisades fuel pool and Action A.2.2 is eliminated.

B. LCO 3.7.16, Spent Fuel Assembly Storage.

1. Change the LCO statement from:

   *"The combination of initial enrichment and burnup of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1."*

   to:

   *"The combination of initial enrichment, burnup, and decay time of each irradiated fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1."*

This change adds the decay time of each assembly as an additional requirement for storage in Region II.

2. Similarly, change SR 3.7.16.1 from:

   *"Verify by administrative means that the initial enrichment and burnup of each spent fuel assembly stored in Region II is in accordance with Table 3.7.16-1."*

   to:

   *"Verify by administrative means that the combination of initial enrichment, burnup, and decay time of each irradiated fuel assembly stored in Region II is in accordance with Table 3.7.16-1."*

3. Replace Table 3.7.16-1 with Table 4 from EA-SFP-99-03.

The criticality analyses which are the basis for this license amendment request credit both burnup and decay time as well as boron when showing that the 95/95 $k_{eff}$ is less than 0.95 in the Region II fuel storage racks. Table 4 from EA-SFP-99-03 provides the updated burnup and enrichment combinations that are acceptable for storage in Region II.

C. Specification 4.3, Fuel Storage

1. Change the allowed enrichment in 4.3.1.1.a from:

   *"having a maximum enrichment of 4.40 weight percent"*

   to:

   *"having a maximum planar average U-235 enrichment of 4.95 weight percent."*

2. Change Specification 4.3.1.1.d from:

   *"Assemblies with enrichments above 3.27 weight percent $U_{235}$ must contain 216 rods which are either $UO_2$, $Gd_2O_3UO_2$, or solid metal."*

   to:

   *"New or irradiated fuel assemblies."*

The criticality analyses which are the basis for this license amendment request show that the 95/95 $k_{eff}$ for the Region I fuel storage racks is less than 0.95 assuming the enrichment of an assembly is less than or equal to 4.95 wt% U-235. The design basis assembly is a 216 pin Palisades assembly. Earlier assembly types with less than 216 pins and guide tubes are considered bounded since their maximum enrichment is less than or equal to 3.27 wt%. Hence the calculation bounds all assemblies currently stored at Palisades and those foreseen in the future. Any new designs other than those assumed in the calculation, including but not limited to different numbers of fueled pins, different pellet diameters, and different pellet densities, will need to be evaluated against the design basis calculation before being stored in the racks.

3. Change the allowed enrichment in Specification 4.3.1.2.a from:

   "having a maximum enrichment of 3.27 weight percent"

   to:

   *"having a maximum planar average* U-235 *enrichment of 4.60 weight percent."*

4. Add a new specification 4.3.1.2.b that states:

   *"$k_{eff}$ < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR."*

5. Renumber existing specification 4.3.1.2.b to 4.3.1.2.c and revise the leading phrase from:

   "$k_{eff} \leq$ 0.95 if fully flooded with unborated water,"

   to:

   *"$k_{eff} \leq$ 0.95 if fully flooded with water borated to 850 ppm,"*

6. Renumber Specifications 4.3.1.2.c and 4.3.1.2.d. Change Specification 4.3.1.2.e (former 4.3.1.2.d) from:

   *"New or partially spent fuel assemblies which meet the initial enrichment and burnup requirements of Table 3.7.16-1."*

   to:

   *"New or irradiated fuel assemblies which meet the initial enrichment, burnup, and decay time requirements of Table 3.7.16-1."*

   The criticality analyses which are the basis for this license amendment show that the 95/95 $k_{eff}$ for the Region II fuel storage racks is less than 0.95 assuming the enrichment of an assembly is less than or equal to 4.60 wt% U-235 and assuming 850 ppm boron in the pool water. The analyses also ensure $k_{eff}$ < 1.0 assuming 0.0 ppm boron. Table 3.7.16--1 as revised in this amendment contains the burnup, enrichment and decay time combinations shown acceptable in EA-SFP-99-03.

7. Change the allowed enrichment in 4.3.1.3.a from:

   *"Fuel assemblies having a maximum average planar $U_{235}$ enrichment of 4.20 weight percent"*

   to:

   *"Twenty-four unirradiated fuel assemblies having a maximum planar average* U-235 *enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3.-1, or*

   *Thirty-six unirradiated fuel assemblies having a maximum planar average* U-235 *enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure. 4.3.-1."*

8. Delete existing Specification 4.3.1.3.c

9. Renumber existing Specification 4.3.1.3d to 4.3.1.3c

10. Add a new figure, Figure 4.3-1; Figure 3 from EA-SFP-99-03.

The criticality analyses which are the basis for this license amendment show that the 95/95 $k_{eff}$ for the new fuel storage rack is less than 0.95 assuming enrichment up to 4.05 wt% U-235 when fully loaded with 36 un-irradiated assemblies. The analyses also show the 95/95 $k_{eff}$ for the new fuel storage rack is less than 0.95 when loaded with only 24 un-irradiated assemblies with enrichment up to 4.95 wt% U-235. The center row of the rack is left empty under this configuration. Figure 3 from EA-SFP-99-03 provides a graphical description of both loading patterns. The figure shows ½ of the new fuel storage rack which is symmetrical about the axis shown. The design basis assembly is a 216 pin Palisades assembly. Earlier assembly types with less than 216 pins and guide tubes are considered bounded since their enrichment is less than or equal to 3.27 wt%. More importantly, all assemblies with less than 216 pins have been irradiated and cannot be stored in the new fuel storage racks. Any new designs other than that assumed in the calculation, including but not limited to different numbers of fueled pins, different pellet diameters, and different pellet densities, will need to be evaluated against the design basis calculation before being stored in the racks.

D. Bases Changes

Corresponding changes to the Bases for Sections B 3.7.15 and B 3.7.16 will be made upon approval of this request. Copies of the revised and marked-up Bases pages are included in the attachments for reviewer information.

## IV. <u>Determination of No Significant Hazards Consideration</u>

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10 CFR 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

### A Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

There is no increase in the probability of the accidental misloading of irradiated fuel assemblies into the spent fuel pool racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification spent fuel rack storage configuration limitations.

There is no increase in the consequences of the accidental misloading of irradiated fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. The proposed Technical Specifications limitations will ensure that an adequate spent fuel pool boron concentration will be maintained.

There is no increase in the probability of a fuel assembly drop accident in the spent fuel pool when considering the presence of soluble boron in the spent fuel pool water for criticality control. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water. The criticality analysis showed the reactivity increase associated with a fuel assembly drop accident in the spent fuel pool is bounded by the misloading accident.

There is no increase in the probability or consequences of the loss of normal cooling to the spent fuel pool water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water.

The criticality analyses documented in EA-SFP-99-03, "*Palisades New Fuel Storage, Fuel Pool and Fuel Handling Criticality Safety Analysis*", show at a 0.95 probability and a 95% confidence level (95/95) that $k_{eff}$ is less than 0.95 under all normal and credible accident conditions. Therefore, the consequences of accidents previously evaluated are not increased.

The Boraflex neutron absorber panels that are present in the Region II fuel storage racks were not credited in the criticality calculations for the postulated accidents. This assumption would cause the positive reactivity addition expected by any of the postulated accidents to increase. However, the additional negative reactivity provided by the proposed 1350 ppm boron concentration limit for accident conditions will compensate for the increased reactivity which would result. The 1350 ppm accident requirement is 500 ppm above the 850 ppm concentration required by proposed specification 4.3.1.2.c. The use of the double contingency principle along with the requirements imposed by LCO 3.9.1, LCO 3.7.15, SR 3.7.15.1 and SR 3.9.1.1 will ensure that adequate soluble boron will be maintained in the spent fuel pool water at all times. Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

**B Does the change create the possibility of a new or different kind of accident from any previously evaluated?**

Spent fuel handling accidents have been analyzed in Section 14.11 and 14.19 of the Final Safety Analysis Report.

Criticality accidents in the spent fuel pool have been analyzed in the previous criticality evaluations which are the bases for the present Technical Specifications.

The existing Palisades Technical Specifications allow storage of 4.2 wt% and 4.4 wt% enriched assemblies in the new fuel storage and Region I fuel storage racks respectively. The possibility of placing an assembly of greater enrichment than allowed in either of these racks exists today. Changing the allowed enrichments does not create a new or different kind of accident.

The existing Palisades Technical Specifications contain limitations on the spent fuel pool boron concentration. Current Specification 3.7.15, which covers the storage of fuel assemblies in an unverified condition, contains a requirement for spent fuel pool boron concentration to be >= 1720 ppm. The actual boron concentration in the spent fuel pool has typically been kept at a higher value for refueling purposes. Proposed Specification 3.7.15 establishes new boron concentration requirements for the spent fuel pool water consistent with the requirements established by the new criticality analysis (EA-SFP-99-03, "*Palisades New Fuel Storage, Fuel Pool and Fuel Handling Criticality Safety Analysis*"). Since soluble boron has always been maintained in the spent fuel pool water, and is currently required by Technical Specifications under some circumstances, the implementation of this new requirement to compensate for not taking credit for the Boraflex contained in the Region II fuel storage racks and to allow an increase in the enrichment limit for all storage locations will have little effect on normal pool operations and maintenance.

Because soluble boron has always been present in the spent fuel pool and is required by current Technical Specifications as discussed above, a dilution of the spent fuel pool soluble boron has always been a possibility. However, it was shown EA-SFP-99-03 that a dilution of the Palisades spent fuel pool which could increase the rack $k_{eff}$ to greater than 0.95 is not a credible event because it would be recognized and terminated well before the limit was reached. Therefore, the implementation of new limitations on the spent fuel pool boron concentration will not result in the possibility of a new kind of accident.

Revised Specification 4.3.1.2 continues to specify the requirements for acceptable storage of assemblies in the Region II fuel storage racks. While the proposed revision changes the burnup requirements as a function of enrichment and creates decay time requirements, the possibility of misloading Region II is not new or different from the possibility present under existing Technical Specifications.

Since the proposed spent fuel spool storage limitations will be similar to those currently in the Palisades Technical Specifications, the new limitations will not have any significant effect on normal spent fuel pool operations and maintenance and will not create the possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the spent fuel pool loading configuration meets specified requirements. There is no significant change in plant configuration, equipment design or equipment.

**C Does this change involve a significant reduction in the margin of safety.**

The Technical Specification changes proposed by this Technical Specifications Change Request, and the resulting spent fuel storage operation limits, will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis found in EA-SFP-99-03, "*Palisades New Fuel Storage, Fuel Pool and Fuel Handling Criticality Safety Analysis*".

EA-SFP-99-03 shows that the Palisades new fuel storage rack loaded under the constraints proposed in this license amendment will have a 95/95 $k_{eff}$ less than 0.95 under all normal and credible abnormal conditions and considering all uncertainties and applicable biases. This margin of safety is consistent with the margin of safety provided by the existing Technical Specifications and their basis calculations.

EA-SFP-99-03 shows that the Palisades Region I fuel storage rack, loaded under the constraints proposed in this license amendment, will have a 95/95 $k_{eff}$ less than 0.95 under all normal and credible abnormal conditions and considering all uncertainties and applicable biases. The analyses of Region I assume 0.0 ppm of boron in the fuel pool water under normal storage conditions and consider the presence of boron during accident configurations as allowed by the double contingency principle. This margin of safety is consistent with the margin of safety provided by the existing Technical Specifications and their basis calculations.

The current Region II criticality design basis discussed in the existing Technical Specifications Section 4.3.1.2 and FSAR Section 9.11.3.2 shows $k_{eff} < 0.95$ assuming 0.0 ppm boron in the water and credit for assembly burnup. Region II criticality calculations described in EA-SFP-99-03 credit both boron and fuel burnup to show that the 95/95 $k_{eff}$ is less than 0.95. Because boron is assumed, there is a theoretical reduction in the margin of safety. Recently the NRC has approved and documented the requirements for use of soluble boron in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998. The credit is allowed based on the ability to show that an acceptable margin between actual and required boron concentrations are present, that adequate controls are in place to verify the boron concentrations remain acceptable, and that an evaluation of any potential boron dilution of the pool shows that the dilution will be terminated prior to diluting below the criticality limit. Therefore the precedent exists for the use of soluble boron credit in the spent fuel pool for criticality control during normal storage scenarios.

There is considerable margin between the boron concentration required to keep the 95/95 $k_{eff}$ below 0.95 and the minimum boron concentration allowed by Technical Specifications. Under the proposed license amendment, spent fuel boron concentration remains controlled by Technical Specifications and is required to be greater than 1720 ppm at all times. Additionally, the requirement increases to the Refueling boron concentration (typically between 2500 and 3000 ppm) whenever the Plant is in Mode 6.

The Technical Specification surveillance for verifying boron concentration while in Mode 6 is every 72 hours. The frequency for verifying boron concentration in the other modes is weekly. These frequencies, combined with the administrative controls in place, ensure that the boron requirements are maintained.

Finally, a boron dilution analysis was performed. Eleven different dilution scenarios were identified and evaluated. The time to dilute from the Technical Specification limit for boron of 1720 ppm down to the required boron concentration to maintain $k_{eff}$ at or below 0.95 was calculated. The eleven scenarios were broken into 2 categories. Category 1 events were evaluated to determine the minimum time available for an operator to recognize and terminate a dilution event before challenging the 850 ppm assumed in the criticality design basis. Category 2 events were evaluated to ensure that Palisades emergency procedures are aligned to allow the operator a recovery path under accident conditions which might leave spent fuel pool level reduced. The Category 2 event starts with a severe loss of water from the spent fuel pool resulting in the uncovering of the cooling system suction piping and a serious loss of shielding water. Procedures currently direct that if non-borated water is all that is available, it should be used to recover minimum shielding water levels and continue to the point that normal cooling can be restored. Procedure changes and enhancements were identified to ensure that a criticality event cannot occur for a category 2 event.

The limiting category 1 scenario involves the addition of pure (0 ppm boron) water from the fire hose station at the 649' elevation, about 40 feet northeast of the spent fuel pool. This station is available for manual use in fighting fires in the pool area. Any dilution scenario involving this fire hose station is bounded by evaluating the direct placement of the 1 ½ inch hose into the pool. A conservative dilution flow rate of approximately 210 gpm is determined by minimizing any flow losses. It would take 123,007 gallons of demineralized water to bring the pool from 1720 ppm down to 850 ppm. Therefore the time to dilute the pool to 850 ppm at a dilution flow rate of 210 gpm would be 9.8 hours.

The first 15,000 gallons of dilution water would fill the pool to its overflow level. The remaining 107,600 gallons needed to achieve the dilution limit would all be over boarded onto the pool deck and down the equipment hatch, elevator shaft, or the stair well, all of which are located within 4 to 10 feet of the pool. The resulting water distribution throughout the auxiliary building and safeguards room basement would result in high sump level alarms in the control room well before the 9.8 hour point. The initiation of the fire system will also result in control room alarms that will lead to auxiliary operators being sent to the area. The large amounts of water on the floor would be easily spotted by the operators whether they have specifically been sent there in response to an alarm or if they were making normal rounds through the auxiliary building and fuel pool on a once per shift basis to complete surveillance procedures DWO-1, and SHO-1. Therefore it is reasonable to assume that the operators will recognize and terminate this event in less than 9.8 hours and that the boron concentration in the spent fuel pool will not drop below 850 ppm.

The dilution event described in Section 14.3 of the FSAR requires that the shutdown margin in the reactor core be sufficient to allow plant operators to recognize and stop any dilution event within 30 minutes while in Mode 6, and within 15 minutes for Modes 1-5. Therefore it is very reasonable to have a pool margin to criticality such that the operators can recognize and stop a dilution within 9.8 hours. There are several physical signs that would clearly lead an operator to stop the limiting dilution event which results in water being over boarded from the spent fuel pool. These signs include the safeguards room sump alarm, the fire system startup alarms, and a significant amount of water on the floor in the immediate vicinity of the pool. Since it is unreasonable to assume that the operators would ignore all of these physical indicators, and since each indicator would be present well before the 9.8 hour limit, then these physical indicators are considered a reliable and adequate means of ensuring the dilution event is terminated in a timely manner.

There is also a key difference which shows that the pool dilution event would provide a much smaller challenge to criticality than the FSAR event. Unlike the FSAR evaluation, even in the unlikely event that the dilution of the spent fuel pool is not stopped and the pool is allowed to dilute down to 0 ppm there would be no safety consequences because the calculations have shown for all storage configurations or fuel handling operations that $k_{eff}$ will not exceed 1.0 even with no soluble boron present.

Because the limiting dilution event will be terminated by operators before reaching the 850 ppm which guarantees a 95/95 $k_{eff} < 0.95$ and because dilution to 0.0 ppm is shown to result in a $k_{eff} < 1.0$, the margin of safety provided by the proposed credit for boron criticality analysis is not significantly reduced from that provided in the current licensing basis.

## V. <u>Conclusion</u>

Based on the evaluation above, and pursuant to 10 CFR 50, Section 50.91, Consumers Energy Company has determined that operation of the Palisades Nuclear Generating Plant in accordance with the proposed license amendment request involves no significant hazards considerations as defined by NRC regulations in 10 CFR 50, Section 50.92

# ENCLOSURE 1
# ATTACHMENT 1

**CONSUMERS ENERGY COMPANY**
**PALISADES PLANT**
**DOCKET 50-255**

**SPENT FUEL POOL BORON CONCENTRATION**

**PROPOSED TECHNICAL SPECIFICATIONS AND BASES PAGES**

13 Pages

## 3.7 PLANT SYSTEMS

### 3.7.15 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.15 The SFP boron concentration shall be ≥ 1720 ppm.

APPLICABILITY: When fuel assemblies are stored in the Spent Fuel Pool.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

---

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. SFP boron concentration not within limit. | A.1 Suspend movement of fuel assemblies in the SFP.<br><br><u>AND</u><br><br>A.2 Initiate action to restore SFP boron concentration to within limit. | Immediately<br><br><br><br>Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.15.1 | Verify the SFP boron concentration is within limit. | 7 days |

## 3.7 PLANT SYSTEMS

### 3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 The combination of initial enrichment, burnup, and decay time of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.
------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 Initiate action to move the noncomplying fuel assembly from Region II. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.16.1 | Verify by administrative means the combination of initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Table 3.7.16-1. | Prior to storing the fuel assembly in Region II |

TABLE 3.7.16-1 (page 1 of 1)

Spent Fuel Minimum Burnup and Decay Requirements
for Storage in Region II of the Spent Fuel Pool and North Tilt Pit

| Initial Enrichment (Wt%) | Burnup (GWD/MTU) No Decay | Burnup (GWD/MTU) 1 Year Decay | Burnup (GWD/MTU) 3 Year Decay | Burnup (GWD/MTU) 5 Year Decay | Burnup (GWD/MTU) 8 Year Decay |
|---|---|---|---|---|---|
| ≤ 1.14 | 0 | 0 | 0 | 0 | 0 |
| >1.14 | 3.477 | 3.477 | 3.477 | 3.477 | 3.477 |
| 1.20 | 3.477 | 3.477 | 3.477 | 3.477 | 3.477 |
| 1.40 | 7.951 | 7.844 | 7.464 | 7.178 | 6.857 |
| 1.60 | 11.615 | 11.354 | 10.768 | 10.319 | 9.847 |
| 1.80 | 14.936 | 14.535 | 13.767 | 13.187 | 12.570 |
| 2.00 | 18.021 | 17.502 | 16.561 | 15.875 | 15.117 |
| 2.20 | 21.002 | 20.417 | 19.313 | 18.499 | 17.611 |
| 2.40 | 23.900 | 23.201 | 21.953 | 21.034 | 20.050 |
| 2.60 | 26.680 | 25.905 | 24.497 | 23.487 | 22.378 |
| 2.80 | 29.388 | 28.528 | 27.006 | 25.879 | 24.678 |
| 3.00 | 32.044 | 31.114 | 29.457 | 28.243 | 26.942 |
| 3.20 | 34.468 | 33.457 | 31.698 | 30.397 | 29.008 |
| 3.40 | 36.848 | 35.783 | 33.920 | 32.544 | 31.079 |
| 3.60 | 39.152 | 38.026 | 36.059 | 34.615 | 33.077 |
| 3.80 | 41.419 | 40.226 | 38.163 | 36.650 | 35.049 |
| 4.00 | 43.661 | 42.422 | 40.257 | 38.673 | 37.007 |
| 4.20 | 45.987 | 44.684 | 42.415 | 40.778 | 39.028 |
| 4.40 | 48.322 | 46.950 | 44.588 | 42.877 | 41.041 |
| 4.60 | 50.580 | 49.158 | 46.690 | 44.911 | 43.003 |

(a) Linear interpolation between two consecutive points will yield acceptable results.

(b) Comparison of nominal assembly average burnup numbers to these in the table is acceptable if measurement uncertainty is ≤ 10%.

# 4.0 DESIGN FEATURES

## 4.1 Site Location

The Palisades Nuclear Plant is located on property owned by Consumers Energy on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

---

## 4.2 Reactor Core

### 4.2.1 Fuel Assemblies

The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix of zircaloy-4 clad fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution. Poison may be placed in the fuel bundles for long-term reactivity control.

### 4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The Region I fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum planar average U-235 enrichment of 4.95 weight percent;

## 4.0 DESIGN FEATURES

4.3 Fuel Storage
4.3.1 Criticality (continued)

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. A nominal 10.25 inch center to center distance between fuel assemblies with the exception of the single Type E rack which has a nominal 11.25 inch center to center distance between fuel assemblies; and

d. New or irradiated fuel assemblies.

4.3.1.2 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with;

a. Fuel assemblies having maximum planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

e. New or irradiated fuel assemblies which meet the discharge burnup requirements of Table 3.7.16-1.

4.3.1.3 The new fuel storage racks are designed and shall be maintained with:

a. Twenty four unirradiated fuel assemblies having a maximum planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. $K_{eff} \leq 0.95$ when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

## 4.0 DESIGN FEATURES

### 4.3 Fuel Storage

#### 4.3.1 Criticality (continued)

c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

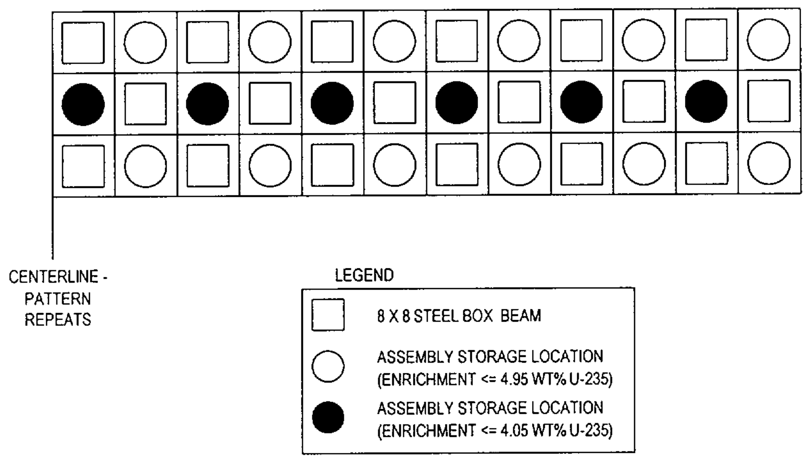
#### 4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

#### 4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.

## 4.0 DESIGN FEATURES




Note: If any assemblies containing fuel enrichments greater than 4.05% U-235 are stored in the New Fuel Storage Rack, the center row must remain empty.

Figure 4.3-1 (page 1 of 1)
New Fuel Storage Rack Arrangement

## B 3.7 PLANT SYSTEMS

## B 3.7.15 Spent Fuel Pool (SFP) Boron Concentration

### BASES

**BACKGROUND**

As described in LCO 3.7.16, "Fuel Assembly Storage," fuel assemblies are stored in the fuel storage racks in accordance with criteria based on initial enrichment, discharge burnup, and decay time.

The criteria were based on the assumption that 850 ppm of soluble boron was present in the spent fuel pool. The pool is required to be maintained at a boron concentration of ≥ 1720 ppm. Criterion 2 of 10 CFR 50.36 (c) (2) requires that criticality control be achieved without credit for soluble boron. However, in 1998 the NRC documented requirements that could be established to maintain criticality below 0.95. This is documented in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998. The precedent of taking credit for soluble boron in spent fuel pool water to provide criticality control has also been established. Soluble boron credit was used in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-14416-NP-A and that methodology was approved for use by an NRC Safety Evaluation dated October 25, 1996. The criteria discussed above was developed using a method that closely followed the Westinghouse methodology. Additionally the requirements specified by the NRC guidance are in place at Palisades.

**APPLICABLE SAFETY ANALYSES**

A fuel assembly could be inadvertently loaded into a fuel storage rack location not allowed by LCO 3.7.16 (e.g., an insufficiently depleted or insufficiently decayed fuel assembly). Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of 10 CFR 50.36(c)(2).

**LCO**

The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the SFP.

**APPLICABILITY**

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

BASES

| | |
|---|---|
| ACTIONS | The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.<br><br>If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.<br><br><u>A.1. and A.2</u><br><br>When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. |
| SURVEILLANCE REQUIREMENTS | <u>SR 3.7.15.1</u><br><br>This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time. |
| REFERENCES | None |

## B 3.7 PLANT SYSTEMS

## B 3.7.16 Fuel Assembly Storage

BASES

---

BACKGROUND

The fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 fuel assemblies, which includes storage for failed fuel canisters. The fuel storage racks are grouped into two regions, Region I and Region II per Figure 3.7.16-1. The racks are designed as a Seismic Category I structure able to withstand seismic events. Region I contains racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single rack in the north tilt pit having an 11.25 inch by 10.69 inch center-to-center spacing. Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and poison concentration, Region II racks have more limitations for fuel storage than Region I racks. Further information on these limitations can be found in Section 4.0, "Design Features." These limitations (e.g., enrichment, burnup) are sufficient to maintain a $k_{eff}$ of $\leq 0.95$ for fuel of original enrichment of up to 4.95% for Region I, and 4.6% for Region II.

---

APPLICABLE SAFETY ANALYSES

The fuel storage facility was originally designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans. The current criticality calculations also take credit for soluble boron to prevent criticality.

The spent fuel assembly storage meets the requirements specified in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998." This document established the requirements for use of soluble boron to maintain $k_{eff}$ below 0.95.

The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36(c)(2).

---

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the $k_{eff}$ of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with water, borated to 850 ppm. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

---

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

---

BASES

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the combination of initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO prior to placing the fuel assembly in a Region II storage location.

REFERENCES

None

BASES

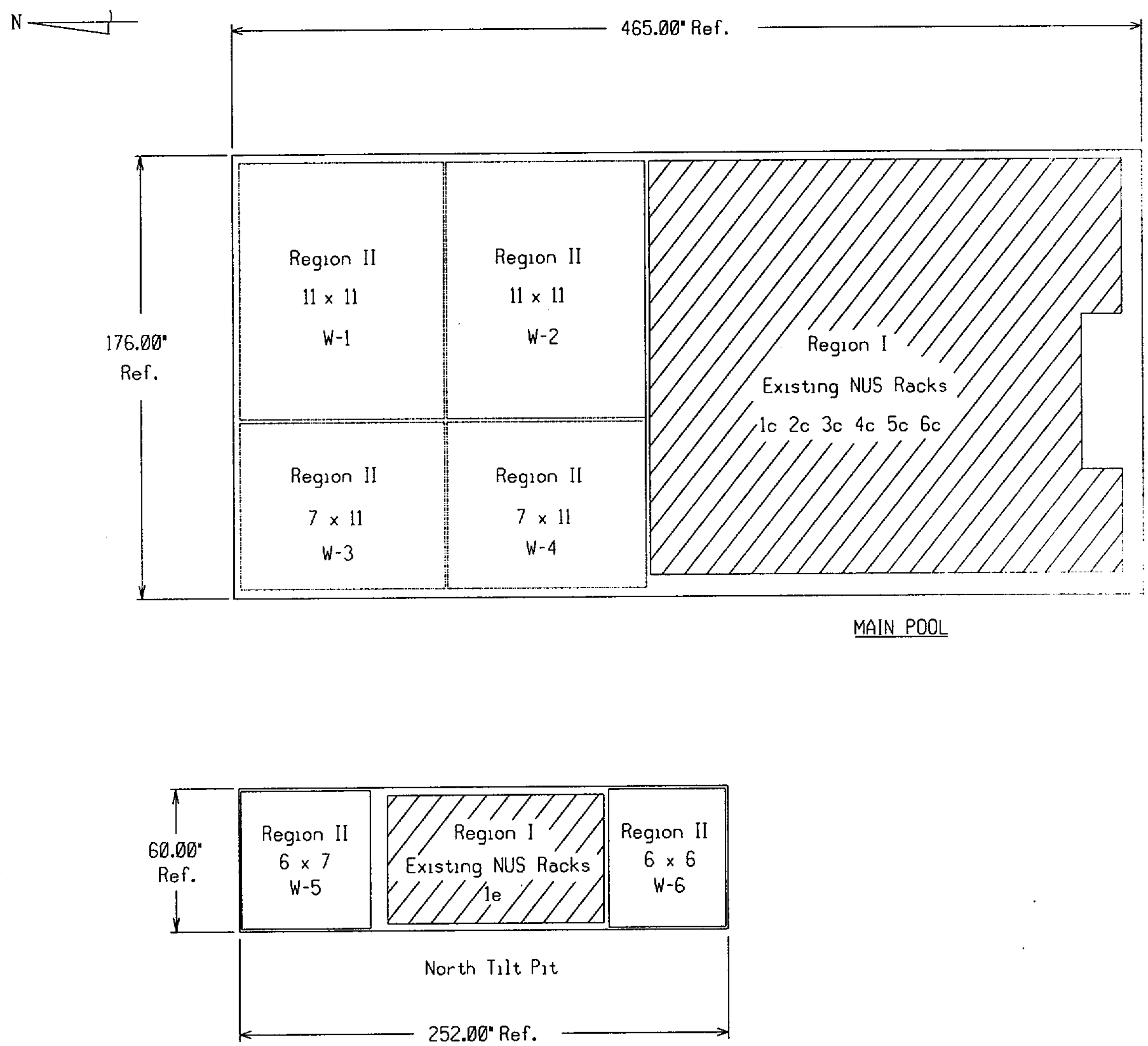



Figure B 3.7.16-1 (page 1 of 1)
Spent Fuel Pool Arrangement

# ENCLOSURE 1
# ATTACHMENT 2

**CONSUMERS ENERGY COMPANY**
**PALISADES PLANT**
**DOCKET 50-255**

**SPENT FUEL POOL BORON CONCENTRATION**

**EXISTING PAGES MARKED TO SHOW PROPOSED CHANGES**

14 Pages

## 3.7 PLANT SYSTEMS

### 3.7.15 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.15 The SFP boron concentration shall be ≥ 1720 ppm.

APPLICABILITY: When fuel assemblies are stored in the ~~SFP and a verification of the stored assemblies has not been performed since the last movement of fuel assemblies in the SFP~~Spent Fuel Pool.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. SFP boron concentration not within limit. | A.1 Suspend movement of fuel assemblies in the SFP. | Immediately |
| | AND | |
| | A.2~~.1~~ Initiate action to restore SFP boron concentration to within limit. | Immediately |
| | ~~OR~~ | |
| | ~~A.2.2 Initiate action to perform a SFP verification.~~ | ~~Immediately~~ |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.15.1 | Verify the SFP boron concentration is within limit. | 7 days |

# 3.7 PLANT SYSTEMS

## 3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 The combination of initial enrichment, ~~and~~ burnup, and decay time of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

ACTIONS

-----NOTE-----

LCO 3.0.3 is not applicable.

----------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 Initiate action to move the noncomplying fuel assembly from Region II. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.16.1 | Verify by administrative means the combination of initial enrichment, ~~and~~ burnup, and decay time of the fuel assembly is in accordance with Table 3.7.16-1. | Prior to storing the fuel assembly in Region II |

~~TABLE 3.7.16-1 (page 1 of 1)~~

~~Spent Fuel Burnup Requirements~~
~~for Storage in Region II of~~
~~the Spent Fuel Pool and North Tilt Pit~~

| ~~Initial Enrichment in Weight % of $U^{235}$~~ | ~~Discharge Burnup Gigawatt Days per Metric Ton~~ |
|---|---|
| ~~1.5~~ | ~~0~~ |
| ~~1.6~~ | ~~1.9~~ |
| ~~1.8~~ | ~~5.2~~ |
| ~~2.0~~ | ~~8.5~~ |
| ~~2.2~~ | ~~11.5~~ |
| ~~2.4~~ | ~~14.1~~ |
| ~~2.6~~ | ~~16.6~~ |
| ~~2.8~~ | ~~18.8~~ |
| ~~3.0~~ | ~~20.9~~ |
| ~~3.2~~ | ~~22.9~~ |
| ~~3.27~~ | ~~23.5~~ |

~~Linear interpolation between two consecutive points will yield conservative results.~~

TABLE 3.7.16-1 (page 1 of 1)

Spent Fuel Minimum Burnup and Decay Requirements
for Storage in Region II of the Spent Fuel Pool and North Tilt Pit

| Initial Enrichment (Wt%) | Burnup (GWD/MTU) No Decay | Burnup (GWD/MTU) 1 Year Decay | Burnup (GWD/MTU) 3 Year Decay | Burnup (GWD/MTU) 5 Year Decay | Burnup (GWD/MTU) 8 Year Decay |
|---|---|---|---|---|---|
| ≤ 1.14 | 0 | 0 | 0 | 0 | 0 |
| >1.14 | 3.477 | 3.477 | 3.477 | 3.477 | 3.477 |
| 1.20 | 3.477 | 3.477 | 3.477 | 3.477 | 3.477 |
| 1.40 | 7.951 | 7.844 | 7.464 | 7.178 | 6.857 |
| 1.60 | 11.615 | 11.354 | 10.768 | 10.319 | 9.847 |
| 1.80 | 14.936 | 14.535 | 13.767 | 13.187 | 12.570 |
| 2.00 | 18.021 | 17.502 | 16.561 | 15.875 | 15.117 |
| 2.20 | 21.002 | 20.417 | 19.313 | 18.499 | 17.611 |
| 2.40 | 23.900 | 23.201 | 21.953 | 21.034 | 20.050 |
| 2.60 | 26.680 | 25.905 | 24.497 | 23.487 | 22.378 |
| 2.80 | 29.388 | 28.528 | 27.006 | 25.879 | 24.678 |
| 3.00 | 32.044 | 31.114 | 29.457 | 28.243 | 26.942 |
| 3.20 | 34.468 | 33.457 | 31.698 | 30.397 | 29.008 |
| 3.40 | 36.848 | 35.783 | 33.920 | 32.544 | 31.079 |
| 3.60 | 39.152 | 38.026 | 36.059 | 34.615 | 33.077 |
| 3.80 | 41.419 | 40.226 | 38.163 | 36.650 | 35.049 |
| 4.00 | 43.661 | 42.422 | 40.257 | 38.673 | 37.007 |
| 4.20 | 45.987 | 44.684 | 42.415 | 40.778 | 39.028 |
| 4.40 | 48.322 | 46.950 | 44.588 | 42.877 | 41.041 |
| 4.60 | 50.580 | 49.158 | 46.690 | 44.911 | 43.003 |

(a) Linear interpolation between two consecutive points will yield acceptable results.

(b) Comparison of nominal assembly average burnup numbers to these in the table is acceptable if measurement uncertainty is ≤ 10%.

# 4.0 DESIGN FEATURES

## 4.1 Site Location

The Palisades Nuclear Plant is located on property owned by Consumers Energy on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

## 4.2 Reactor Core

### 4.2.1 Fuel Assemblies

The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix of zircaloy-4 clad fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution. Poison may be placed in the fuel bundles for long-term reactivity control.

### 4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The Region I fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum planar average U-235 enrichment of ~~4.40~~ 4.95 weight percent;

# 4.0 DESIGN FEATURES

## 4.3 Fuel Storage

4.3.1 Criticality (continued)

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. A nominal 10.25 inch center to center distance between fuel assemblies with the exception of the single Type E rack which has a nominal 11.25 inch center to center distance between fuel assemblies; and

d. New or ~~partially spent~~ irradiated fuel assemblies. ~~Assemblies with enrichments above 3.27 weight percent $U_{235}$ must contain 216 rods which are either $UO_2$, $Gd_2O_3$-$UO_2$ or solid metal.~~

4.3.1.2 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with;

a. Fuel assemblies having maximum planar average U-235 enrichment of ~~3.27~~ 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

~~b~~c. $K_{eff} \leq 0.95$ if fully flooded with ~~unborated~~ water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

~~c~~d. A nominal 9.17 inch center to center distance between fuel assemblies; and

~~d~~e. ~~Partially spent~~ New or irradiated fuel assemblies which meet the discharge burnup requirements of Table 3.7.16-1.

4.3.1.3 The new fuel storage racks are designed and shall be maintained with:

a. ~~Fuel~~ Twenty four unirradiated fuel assemblies having a maximum ~~average planar $U_{235}$~~ planar average U-235 enrichment of ~~4.20~~ 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. $K_{eff} \leq 0.95$ when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

# 4.0 DESIGN FEATURES

## 4.3 Fuel Storage

### 4.3.1 Criticality (continued)

~~c. Assemblies must contain 216 rods which are either $UO_2$, $Gd_2O_3$-$UO_2$ or solid metal.~~

~~d~~c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

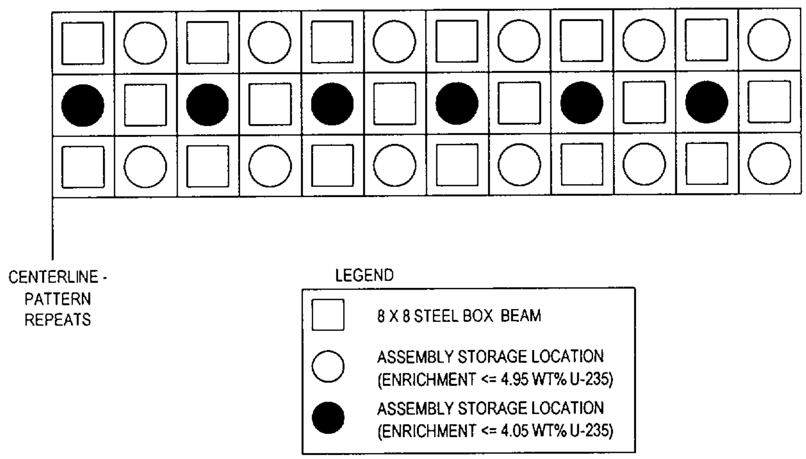
### 4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

### 4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.

## 4.0 DESIGN FEATURES




Note: If any assemblies containing fuel enrichments greater than 4.05% U-235 are stored in the New Fuel Storage Rack, the center row must remain empty.

Figure 4.3-1 (page 1 of 1)
New Fuel Storage Rack Arrangement

# B 3.7 PLANT SYSTEMS

## B 3.7.15 Spent Fuel Pool (SFP) Boron Concentration

BASES

**BACKGROUND**

As described in LCO 3.7.16, "~~Spent~~ Fuel Assembly Storage," fuel assemblies are stored in the ~~spent~~ fuel storage racks in accordance with criteria based on initial enrichment, ~~and~~ discharge burnup, and decay time. ~~Although the water in the spent fuel pool is normally borated to ≥ 1720 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron.~~

The criteria were based on the assumption that 850 ppm of soluble boron was present in the spent fuel pool. The pool is required to be maintained at a boron concentration of ≥ 1720 ppm. Criterion 2 of 10 CFR 50.36 (c) (2) requires that criticality control be achieved without credit for soluble boron. However, in 1998 the NRC documented requirements that could be established to maintain criticality below 0.95. This is documented in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998. The precedent of taking credit for soluble boron in spent fuel pool water to provide criticality control has also been established. Soluble boron credit was used in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-14416-NP-A and that methodology was approved for use by an NRC Safety Evaluation dated October 25, 1996. The criteria discussed above was developed using a method that closely followed the Westinghouse methodology. Additionally the requirements specified by the NRC guidance are in place at Palisades.

**APPLICABLE SAFETY ANALYSES**

A fuel assembly could be inadvertently loaded into a ~~spent~~ fuel storage rack location not allowed by LCO 3.7.16 (e.g., an ~~unirradiated fuel assembly or an~~ insufficiently depleted or insufficiently decayed fuel assembly). ~~This accident is analyzed assuming the extreme case of completely loading the fuel pool racks with unirradiated assemblies of maximum enrichment~~. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of 10 CFR 50.36(c)(2).

**LCO**

The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the SFP.

**APPLICABILITY**

This LCO applies whenever fuel assemblies are stored in the spent fuel pool ~~until a complete spent fuel pool verification of the stored assemblies has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.~~

BASES

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

~~A.1, A.2.1, and A.2.2~~
A.1, and A.2

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. ~~Alternately, beginning a verification of the SFP fuel locations to ensure proper locations of the fuel can be performed.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

None

# B 3.7 PLANT SYSTEMS

## B 3.7.16 ~~Spent~~ Fuel Assembly Storage

BASES

| | |
|---|---|
| BACKGROUND | The ~~spent~~ fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 ~~irradiated~~ fuel assemblies, which includes storage for failed fuel canisters. The ~~spent~~ fuel storage racks are grouped into two regions, Region I and Region II per Figure 3.7.16-1. The racks are designed as a Seismic Category I structure able to withstand seismic events. Region I contains racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single rack in the north tilt pit having an 11.25 inch by 10.69 inch center-to-center spacing. Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and poison concentration, Region II racks have more limitations for fuel storage than Region I racks. Further information on these limitations can be found in Section 4.0, "Design Features." These limitations (e.g., enrichment, burnup) are sufficient to maintain a $k_{eff}$ of $\leq$ 0.95 for ~~spent~~ fuel of original enrichment of up to ~~4.40%.~~ 4.95% for Region I, and 4.6% for Region II. |
| APPLICABLE SAFETY ANALYSES | The ~~spent~~ fuel storage facility ~~is~~ was originally designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans. The current criticality calculations also take credit for soluble boron to prevent criticality.<br><br>The spent fuel assembly storage meets the requirements specified in "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998." This document established the requirements for use of soluble boron to maintain $k_{eff}$ below 0.95.<br><br>The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36(c)(2). |
| LCO | The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the $k_{eff}$ of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with ~~unborated~~ water, borated to 850 ppm. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1. |
| APPLICABILITY | This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit. |

BASES

ACTIONS

~~Required Action A.1 is~~ The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

~~A.1~~

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the combination of initial enrichment, ~~and~~ burnup, and decay time of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO prior to placing the fuel assembly in a Region II storage location.

REFERENCES

None

BASES

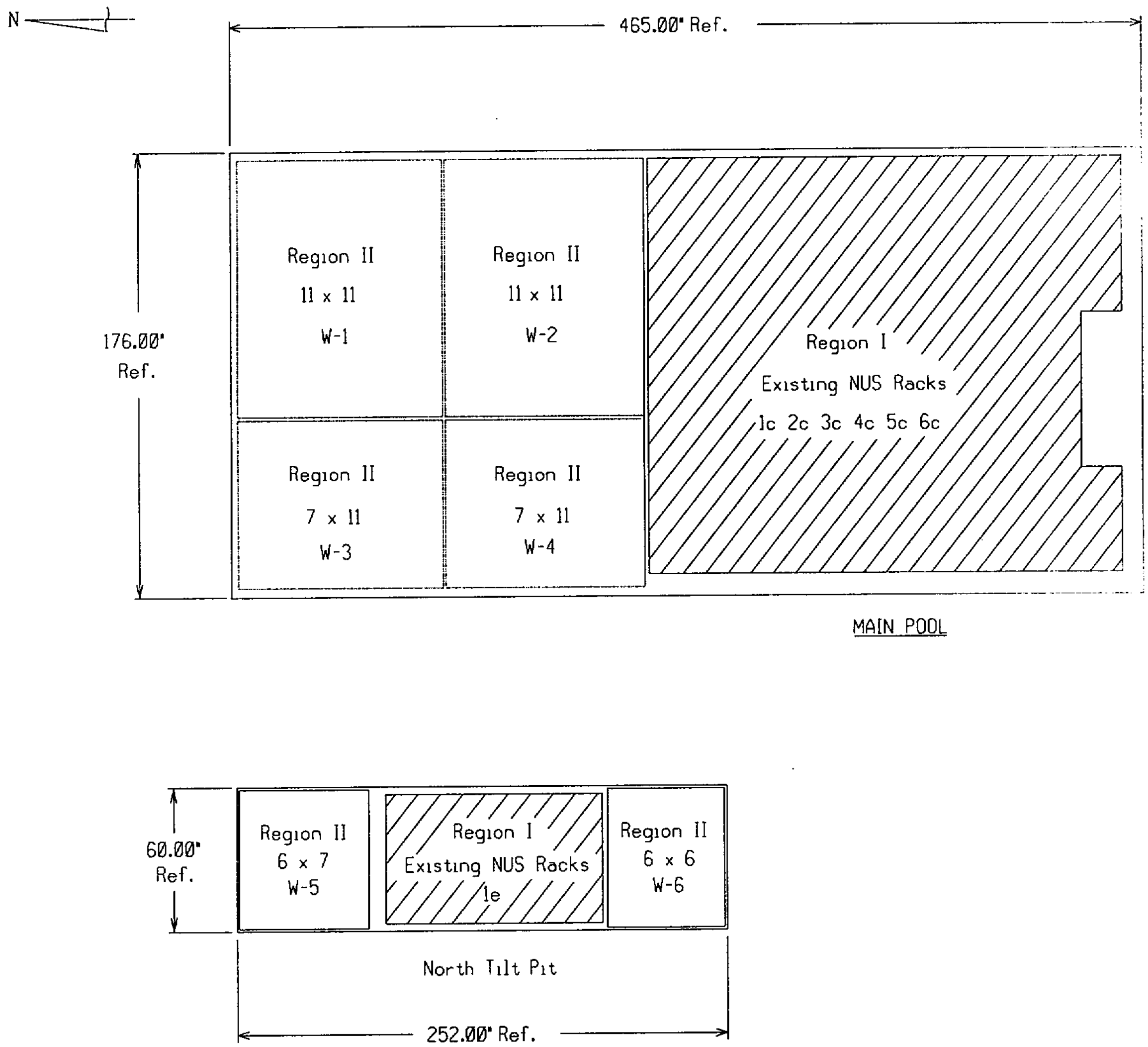



Figure B 3.7.16-1 (page 1 of 1)
Spent Fuel Pool Arrangement

# ENCLOSURE 2

**CONSUMERS ENERGY COMPANY**
**PALISADES PLANT**
**DOCKET 50-255**

**TECHNICAL SPECIFICATIONS CHANGE REQUEST**
**SPENT FUEL POOL BORON CONCENTRATION**

**ENGINEERING ANALYSIS EA-SFP-99-03**

54 Pages

Consumers Energy

EA-SFP-99-03

PALISADES NUCLEAR PLANT
**ENGINEERING ANALYSIS COVER SHEET**

Total Pages 54

Palisades New Fuel Storage, Fuel Pool and Fuel Handling Criticality Safety Analysis.

INITIATION AND REVIEW

| Calculation Status | Preliminary ☐ | Pending ☒ | Final ☒ 2/13/01 | Superseded ☐ | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Rev. | Description | Initiated | | Init. Appd. By | Review Method | | | Technically Reviewed | | Revr. Appd. By | SDR Appd. By |
| | | By | Date | | Alternate Calc | Detailed Review | Qual Test | By | Date | | |
| 0 | Original Issue | T.C. Duffy / R.D. Radulovich | 10/23/00 / 10/23/2000 | | | X | | G.E. Jarka | 10/24/00 | | |
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## EXECUTIVE SUMMARY


This report fully documents the criticality safety analysis for the Palisades new fuel storage, fuel pool storage and fuel inspection and transfer machinery. Palisades dry fuel storage facilities are **NOT** included in the scope of this report. Three specific objectives are accomplished:

1. The new fuel storage racks are reanalyzed to allow storage of fresh fuel up to 4.95 wt% maximum planar average enrichment.

2. The fuel pool is reanalyzed to allow storage or handling of :
   a. Fresh fuel up to 4.95 wt% maximum planar average enrichment in Region I storage racks.
   b. Fresh fuel up to 1.14 wt% maximum planar average enrichment and burned fuel of equivalent reactivity up to 4.6 wt% initial maximum planar average enrichment in Region II racks.
   c. Fresh or burned fuel up to 4.95 wt% maximum planar average enrichment in the fuel elevator and transfer machine.

The Region II fuel storage rack evaluation conservatively ignores the Boraflex poison material. Assembly burnup and subsequent decay time is considered in the Region II calculations. No credit for assembly burnup is taken when evaluating the new fuel storage racks, Region I racks, fuel elevator or the transfer machine. The presence of 1350 ppm soluble boron is credited to ensure that $k_{eff}$ remains below 0.95 under all normal and credible accident conditions in the fuel pool.

# TABLE OF CONTENTS

Section: Page:

## 1.0 INTRODUCTION

Palisades is continually looking to design fuel cycles which are as economical as possible. Many factors such as, pellet density, assembly enrichment and desired cycle length go into determining the most cost effective fuel design. Recent reloads have seen an increase in the reactivity of the fresh assembly. The spent fuel pool and new storage pit criticality analyses restrict the reactivity of fresh fuel because the maximum analyzed enrichment is 4.4 and 4.2 wt% respectively. Region II is currently limited to fuel assemblies which have an initial *maximum planar average* enrichment less than or equal to 3.7 wt%. Recent fuel designs have exceeded this limit, and hence, cannot be placed in Region II under the Technical Specifications [1].

The objective is to reanalyze the Palisades new fuel storage, fuel pool, fuel elevator and transfer machine using assembly design assumptions which bound fuel in the foreseeable future. This includes increased pellet density and diameter, higher enrichments, and longer stack height than fuel which has previously been analyzed.

## 1.1 THE PALISADES FACILITIES

The layout of the Palisades fuel storage facilities is shown in Figures 1 & 2. The fuel pool consists of the main pool area and the north tilt pit area. The north tilt pit was originally constructed in anticipation of a second PWR unit on the Palisades site. It is connected to the main pool through a gate which has been removed and is permanently flooded allowing fuel storage. Region I of the pool consists of racks manufactured by NUS while Region II racks were manufactured by Westinghouse. The Region I racks are located at the south end of the main pool and in the center of the north tilt pit. Region II racks occupy the north end of the pool and surround the Region I rack in the north tilt pit. A two inch minimum separation exists between Region I and Region II rack modules. In addition, a new fuel storage rack exists adjacent to the fuel pool. A fuel elevator used to lower fresh fuel into the pool is located at the very south end of the pool. The south tilt pit is separated from the fuel pool by a gate during normal operation. When refueling, the gate is removed connecting the fuel pool, tilt pit and reactor cavity. The transfer machine located in the tilt pit moves assemblies between the fuel pool and the reactor cavity.

### 1.1.1 New Fuel Storage Racks

Although there are seventy-two locations in the new fuel storage racks, only 36 are currently available to store fuel. The other half of the rack locations are occupied either by steel box beams or stainless steel core plugs. A specific loading pattern alternating plugs and fuel is assumed. Such a short loading is necessitated by the regulatory requirement to show sub-criticality ($k_{eff} < 0.95$) at optimum moderation conditions. Original design of the rack allowing storage of 72 assemblies assumed normal dry conditions and did not provide spacing or poisoning to account for increased reactivity with added moderation. This analysis considers two loading patterns. First, thirty-six fresh 4.05 wt% assemblies are considered. The assemblies

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1: Palisades Improved Technical Specifications, Section 4.3.1.2

are positioned in the locations which are not occupied by either core plugs or box beams. A second loading with only twenty-four fresh 4.95 wt% assemblies loaded in the outside two rows of the rack (center row left empty) is also analyzed. Figure 3 shows the two loading patterns considered [2]. Figure 4 shows details the new fuel storage cell design.

### 1.1.2 Region I Fuel Storage Racks

All of the Region I racks were designed and manufactured by NUS. The racks are constructed of stainless steel and have a $B_4C$ neutron poison in all four walls of each storage cell. The majority of Region I racks are located in the main storage pool and have 8.56 inch square storage cells with a 10.25 inch center to center spacing. The north tilt pit pool consists of a north to south row of three storage racks. The center rack is a 10x5 array of storage cells. This rack is designed with a 9.0 inch storage cell inner width in order to store Palisades type control blades as well as fuel assemblies. Figure 5 depicts the Region I main storage pool cell layout.

### 1.1.3 Region II Fuel Storage Racks

Region II racks were designed and manufactured by Westinghouse and have a smaller center-to-center spacing which limits storage to spent fuel and control blades only. Current analysis credit both the presence of a fixed neutron absorber, Boraflex, and assembly burnup when showing a 95/95 $k_{eff}$ below 0.95.

In recent years, questions concerning the long-term Boraflex performance in spent fuel pools have arisen. One specific issue is the radiation-induced shrinkage of Boraflex and the potential for development of tears or gaps [3]. Consumers Energy removed Boraflex surveillance coupons from Region II in August 1993. Significant degradation of the poison material was identified [4]. In response, Blackness testing was performed on a selective sample of Region II rack locations. Results indicated that 36% of the 98 Boraflex panels tested had measurable gaps (½" or wider). The average gap size was 0.7" and the largest gap was 1.0" [5]. The gaps appeared to be distributed randomly in the axial direction. The test concluded that the gaps found were relatively small compared to industry experience. Reactivity effects were considered insignificant. However, Consumers Energy has closely tracked Boraflex degradation by utilization of the RACKLIFE program [6], and close monitoring of silica levels in the spent fuel pool water. The omission

2: Section 3.4 *"Storage Array Description"* and EA-SFP-97-03 provides the rack dimensions used for the analysis.

3: EPRI TR-101986, *"Boraflex Test Results and Evaluation"*.

4: Docket 50-255 License DPR-20, *"Degradation of Boraflex Neutron Absorber in Surveillance Coupons...."*

5: Holtec Report HI-951279, *"Blackness Testing of Boraflex ..."*

6: *"The RACKLIFE Boraflex Rack Life Extension Computer Code: Theory and Numerics"*.

of the Boraflex material from criticality calculations is compensated for by the increase in required assembly burnup and credit for soluble boron in the pool water. Figure 6 shows the Region II rack geometry.

### 1.1.4 Fuel Elevator - Inspection Station

New fuel bundles are transported from the new fuel storage rack to the fuel pool via the fuel elevator. The elevator receives the fuel bundle in its raised position and then travels to the bottom of the fuel pool. The fuel bundle is then picked up by the service platform. The fuel elevator contains an inspection station to allow examination of irradiated fuel. Fuel repairs can be conducted in the elevator/inspection station. Figure 7 depicts the elevator and inspection station as positioned in the south end of the Palisades fuel pool.

### 1.1.5 Transfer Machine (Tilt Machine)

The Fuel Handling System is used to transfer fuel bundles between the refueling cavity and the fuel pool. The refueling machine removes a fuel bundle from the core, transports it to the tilt machine and deposits it in the transfer carriage within the tilt machine. The carriage is then rotated from the vertical position to a horizontal position and moved through the transfer tube to the fuel storage area. The transfer carriage consists of two main structural members which support two fuel assembly cavities and associated bracing. Rollers on one end transfer the load of the carrier and fuel assembly to the track of the tilting machine in the fuel storage area. Figure 8 shows the geometry of the transfer machine assembly cavities.

## 1.2 DESIGN CRITERIA

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 0.95 probability at a 95 percent confidence level that the effective neutron multiplication factor, $k_{eff}$, of the fuel in the rack or handling machine will be less than or equal to 0.95 [7] [8].

Criticality in the Palisades spent fuel pool and new fuel storage rack is prevented by the design of the racks and fuel elevator which provide a minimum separation between fuel assemblies. The Region I fuel pool racks are manufactured with a $B_4C$ neutron poison material in the cell walls. Region II fuel pool racks are manufactured with a Boraflex poison material in the cell walls. As stated earlier, criticality evaluations discussed here conservatively do not consider the Boraflex material. In addition to the minimum spacing between assemblies defined by the construction of the racks and fuel handling machines, credit for soluble boron is taken to ensure $k_{eff} \leq 0.95$ in the Region II racks and fuel transfer machine. Boron is also credited to ensure $k_{eff} \leq 0.95$ under abnormal and accident conditions throughout the pool.

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7: ANSI/ANS 57.2 Section 6.4.2

8: ANSI/ANS 57.3 Section 6.2.4

## 2.0 ANALYTICAL METHODS

The analysis methods employed here follow closely those outlined in the topical report "*Westinghouse Spent Fuel Rack Criticality Analysis Methodology*" [9]. However, different calculation tools are used. The MONK 7A (instead of KENO-Va) Monte Carlo code is used to calculate $k_{eff}$ and CASMO-3 (instead of PHOENIX) is used for reactivity equivalencing. New fuel storage, fuel elevator and transfer machine evaluations employ MONK calculations which consider the "worst case" configuration of the assembly and rack as allowed by manufacturing tolerances. The Region I and Region II fuel pool rack analyses employ MONK calculations at nominal conditions and use CASMO-3 to statistically determine the reactivity effect of variations within manufacturing tolerances. The specifics of each evaluation are highlighted in Sections 3.0 through 6.0 of this report. Sections 2.1, 2.2 and 2.3 discuss the computer codes used, the nominal fuel assembly considered and generic assumptions which apply to all of the criticality evaluations. Section 7.0 addresses possible accident and abnormal conditions.

## 2.1 COMPUTER CODES

MONK is a Monte Carlo neutronics computer code written to assist in the study of criticality safety problems. MONK is distributed and actively supported by AEA Technology with the code development being managed by a collaboration comprising AEA and British Nuclear Fuels [10]. The primary aim of MONK is to calculate the neutron multiplication factor ($k_{eff}$) of systems by the computer simulation of the birth, migration and ultimate fate of a finite sample of typical neutrons. The actual number of neutrons followed or tracked determines the statistical precision associated with the calculated value of $k_{eff}$.

Neutron interactions are considered in the MONK collision processing package call DICE. The standard MONK nuclear data library is a 8220 group library based primarily on UKNDL evaluations. JEF data is used to supplement the library with fission product and higher actinide data not present in UKNDL. This library, together with the point-energy collision processing algorithms, provides a very detailed modeling of the physics. Therefore, the ultimate accuracy of the MONK code largely depends on the numerical accuracy of the basic nuclear data. This continuous energy package has been the subject of extensive validation studies [11]. MONK, when utilizing the UKNDL based nuclear data library, is designed to systematically over predict $k_{eff}$ for uranium oxide systems such as the fuel pool racks and other geometries discussed here. This over prediction is shown in Section 9.0 of this report which discusses the results of code validation and the determination of appropriate biases for the criticality evaluation applications presented.

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9: WCAP-14416-NP-A, "*Westinghouse Spent Fuel Rack Criticality Analysis Methodology*".

10: ANSWERS/MONK(94) 3, MONK Users Guide for Version 7A.

11: ANSWERS/MONK(94)3, MONK Users Guide for Version 7A, Chapter 7, "*Validation*".

CASMO-3 [12] is a multi group two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array. Typical fuel storage rack arrays can also be handled. Nuclear data are collected in a library containing microscopic cross sections in 70 energy groups. Nuclear data are automatically read from the library. The microscopic depletion is calculated in each fuel pin. In the depletion calculation a predictor-corrector approach is used which greatly reduces the number of burnup steps necessary for a given accuracy. CASMO has a user oriented input. Default values are available for many quantities. A 40 group library has been developed and is used in this analysis. Neutron energies cover the range 0 to 10 MeV.

## 2.2 DESIGN BASIS FUEL ASSEMBLY

The fuel assembly is modeled as a 15x15 array of pins. Each assembly contains 216 $UO_2$ rods at the maximum allowed planar average enrichment. Each assembly has eight guide bars and one instrument tube. Fuel parameters given in Table 1 are based on Palisades R-type fuel. Exceptions include enrichment which is artificially increased to 4.95 wt%, pellet density which is increased to 96 %TD and dishing which is completely ignored. These values are considered bounding for all past and future Palisades fuel types as discussed below. Manufacturing tolerances about the nominal values are considered in each of the criticality evaluations described.

## 2.3 ASSUMPTIONS

Three major assumptions apply throughout the criticality evaluations discussed in this analysis. Additional specific assumptions for each calculation (ie Region I, Region II, New Fuel Rack etc ..) are discussed in Sections 3.0 through 6.0. The following paragraphs list the three major assumptions and provide a brief justification for their application.

1) The R-type fuel parameters are assumed. Assembly design includes 216 fueled pins, 8 guide bars, 1 instrument tube. All fueled pins are assumed to be at the assembly *maximum planar average* enrichment. No credit is taken for natural or reduced enrichment axial blankets. No pellet dishing is considered. Fuel assembly structural material such as spacer grids and end fittings are ignored.

Palisades currently has three major sub-groupings of assemblies in its spent fuel pool. All assemblies are a 15x15 array of pins. The original Combustion Engineering fuel contained 212 fueled pins while later Siemens fuel has consisted of either 208 or 216 fueled pins. The 216 pin assembly is shown to be bounding for the conditions found in the fuel pool. The assumption of a average enrichment over all the pins in a 216 pin lattice results in a conservatively high $k_\infty$. In general, the radial enrichment distributions used at Palisades are designed to limit pin peaking in the wide water gaps present in the core. The reduction in pin

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12: SOA-94/9, CASMO-2 Users Manual.

peaking is typically paid for by a decrease in assembly reactivity especially when no burnable absorbers are considered. Reduced axial blanket enrichments typically incorporated into an assembly average enrichment calculation are not used to determine the *maximum planar average* enrichment. In combination, the assumed R-type 216 fueled pin assembly bounds any past Palisades assembly and any design in the foreseeable future [13].

2) Boraflex poison is ignored.

The Boraflex neutron poison material for Region II is conservatively ignored. The physical stability of the Boraflex sheets in the harsh fuel pool environment has been questioned in the past. The elimination of the Boraflex from the calculation conservatively accounts for reactivity effects from any conceivable degradation in the rack poison. This criticality evaluation allows for the reduction in efforts associated with ensuring the continued integrity of the poison in the rack.

3) Soluble boron is conservatively assumed to be comprised of 17% $^{10}B$ atoms and 83% $^{11}B$ atoms.

When soluble boron is considered in the pool water, the $^{10}B$ concentration is artificially reduced. Boron exists naturally as 19.78% $^{10}B$ isotope and 80.2% $^{11}B$ [14]. Calculations discussed in this report conservatively reduce the $^{10}B$ isotope percentage to 17.00% to account for any possible $^{10}B$ depletion and other unknown uncertainties. A 17.00 % $^{10}B$ concentration is considered very conservative since the $^{10}B$ absorption macro - cross section is significantly understated as a result. The $^{10}B$ concentration in the Region I $B_4C$ poison material is calculated according to manufacturing specifications.

## 3.0 NEW FUEL STORAGE RACK 95/95 $k_{eff}$ CALCULATION

The results of the criticality analysis for the Palisades new fuel storage rack are presented in this section [15]. The new fuel storage rack is a 3x24 array of individual assembly locations. Although there are seventy-two locations in the new fuel storage racks, only 36 are currently available to store fuel. The other half of the rack locations are occupied either by steel box beams or stainless steel core plugs. Plugs are placed in alternating locations effectively increasing the minimum separation of any two assemblies stored in the rack. The new fuel storage rack was previously qualified for storage of thirty-six 15x15 fuel assemblies with a maximum enrichment up to 4.20 w/o $^{235}U$ [16]. Figures 3 & 4 show the new fuel storage rack geometry. The calculation presented here shows the 95/95 $k_{eff}$, assuming worst case manufacturing tolerances and optimum moderation, is less than 0.95 when the rack is loaded with thirty-six 4.05 wt% enriched assemblies

13: EA-SFP-98-03, Appendix A.

14: *"Handbook of Chemistry and Physics"*.

15: Details of the calculation and analysis can be found in EA-SFP-97-03.

16: EMF-91-1421(P), *"Criticality Safety Analysis for the Palisades Spent Fuel ...."*

or twenty-four 4.95 wt% assemblies [17]. A specific loading pattern is assumed for each enrichment level (i.e.enrichments $\leq$ 4.05 wt% and enrichments > 4.05 wt% up to 4.95 wt%) as shown in Figure 3. The loading patterns are relatively easy to interpret. Specifically, if any assembly loaded in the new fuel storage rack has a maximum planar average enrichment greater than 4.05 wt%, then no fuel can be loaded into the center row of the rack. The box beams and core plugs remain located in alternating positions regardless of the fuel loading pattern.

## 3.1 METHODS

The MONK code was run to determine the calculated 95/95 $k_{eff}$ for the new fuel storage rack. The MONK calculations took into account the impact of manufacturing tolerance variations on the fuel dimensions and assembly placement in the storage rack. Since the "worst case" manufacturing tolerances and assembly loadings are considered in the MONK calculation, the determination of the 95/95 $k_{eff}$ from the MONK calculated $k_{eff}$ is relatively straight forward. The calculated $k_{eff}$ is added to two times the standard deviation in the MONK calculational results to determine the 95/95 $k_{eff}$.

## 3.2 MAJOR ASSUMPTIONS

In addition to the Assumptions listed in Section 2.3, the following assumptions are used to determine the 95/95 $k_{eff}$ for fuel assemblies stored in the Palisades new fuel storage rack:

1. The new fuel storage rack is modeled assuming the as measured worst case center-to-center cell spacing of 9⅜ inches.
2. Core plugs are modeled as steel box beams.
3. 4.05 wt% calculations assume a staggered loading pattern which allows storage of thirty-six assemblies.
4. 4.95 wt% calculations assume a staggered loading pattern which allows storage of twenty-four assemblies.
5. The rack is assumed flooded with pure (no soluble boron) water at optimum moderator density.

17: The calculations in EA-SFP-97-03 consider assemblies up to a nominal 5.00 wt% maximum planar average enrichment. This enrichment is conservatively reduced to 4.95 wt% to be consistent with other enrichment limits considered in the Region I, elevator and transfer machine evaluations.

## 3.3 CONSERVATIVE ASSEMBLY SPECIFICATIONS

The MONK analysis sets key parameters that describe the fuel assembly and new fuel rack geometry to conservative values within their expected manufacturing tolerances. In general, the cladding is made as thin as possible while the pellet is made as large and as dense as possible. The combination of parameters used to develop the MONK model are presented below and conservatively represent the highest reactivity assembly possible within current manufacturing tolerances.

**Fuel Pellet Density:** A (+) 1.5 % variation about a nominal percent theoretical density of 96.0% is modeled.

**Fuel Pellet Diameter:** A (+) 0.0005 inch variation about the nominal pellet diameter of 0.3600 inches is modeled.

**$^{235}$U Enrichment:** The enrichment tolerance of (+) 0.05 w/o $^{235}$U about the nominal reference enrichments of 5.00 & 4.05 w/o $^{235}$U is modeled [18].

**Cladding ID:** A (-) 0.0015 inch variation about the nominal cladding ID of 0.3670 inches is modeled.

**Cladding OD:** A (-) 0.002 inch variation about the nominal cladding OD of 0.4170 inches is modeled.

## 3.4 STORAGE ARRAY DESCRIPTION

Figure 4 details the new fuel storage rack geometry. The design nominal center-to-center spacing between adjacent storage locations of the new fuel rack is 9½ inches. Measurements have shown a maximum tolerance of ⅛ inch on the "As built" center-to-center spacing. Therefore, the minimum nominal center-to-center separation of 9⅜ inches is used to develop the MONK model. The 8x8 inch structural steel box beams that are placed in alternate storage cells have a nominal wall thickness of $^{5}/_{16}$ inch. A minimum wall thickness of 0.25 inches is used in this analysis. The rack structural material, aluminum, is conservatively modeled. The closest approach of two assemblies is in part limited by a $^{3}/_{16}$ inch thick "L" shaped guide sleeve. These sleeves are located in alternating corners of each cell location and are made of aluminum. Guide bars on the ends of these sleeves are designed to contact the assembly guide bars preventing the fuel pins from contact with the rack structure. Concrete walls are adjacent to three sides of the storage array and are separated from the fuel by 0.5 to 1.5 inches. For the purposes of this analysis, a 16 inch concrete reflector is modeled touching three sides of the storage rack. The top, bottom and fourth side are reflected with water.

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18: The calculations in EA-SFP-97-03 consider assemblies up to a nominal 5.00 wt% maximum planar average enrichment. An additional 0.05 wt% tolerance is considered. This enrichment is conservatively reduced to 4.95 wt% to be consistent with other enrichment limits considered in the Region I, elevator and transfer machine evaluations.

## 3.5 RESULTS

The MONK code is used to calculate the $k_{eff}$ for storage of fuel assemblies in the new fuel storage rack. Two different assembly enrichments are considered each requiring a specific loading pattern to ensure acceptable results. The first case evaluates the storage of thirty-six 4.05 wt% enriched fuel assemblies. The most reactive situation for this case is an offset assembly spacing. The calculated $k_{eff}$ is 0.9470. The 95/95 $k_{eff}$ is developed by adding two times the MONK calculation standard deviation to the MONK calculated value as demonstrated by the equation below. The 95/95 $k_{eff}$ for the storage of thirty-six 4.05 wt% enriched fuel assemblies is 0.9482 [19].

The second case evaluates the storage of twenty-four 4.95 wt% enriched fuel assemblies. The most reactive situation for this case is a centered assembly spacing. The calculated $k_{eff}$ is 0.9349. The 95/95 $k_{eff}$ is again developed by adding two times the MONK calculation standard deviation to the MONK calculated value. The 95/95 $k_{eff}$ for the storage of twenty-four 4.95 wt% enriched fuel assemblies is 0.9361 [20].

$$k_{eff_{95/95}} = k_{MONK} + 2s_{MONK}$$

Where:

| | |
|---|---|
| $k_{eff95/95}$ | is the 95/95 $k_{eff}$ of 4.95 wt% fresh fuel with no Boron. |
| $k_{MONK}$ | is the MONK calculated $k_{eff}$. |
| $2s_{MONK}$ | is 2x the MONK calculation standard deviation = 0.0012 $\Delta k$ |

The results demonstrate that the $k_{eff}$ in the new fuel storage rack, at a 95 percent probability with a 95 % confidence, is below 0.95 considering the worst credible storage array conditions for the following loadings:

1. Twenty-four 4.95 wt% enriched fuel assemblies may be stored in the pattern specified in Figures 3.

2. Thirty-six 4.05 wt% enriched fuel assemblies may be stored in the pattern specified in Figures 3.

The new fuel array is normally dry, with a $k_{eff}$ < 0.6 for either the 4.05 wt% or 4.95 wt% loadings. However, the calculation is performed assuming optimum moderation (rack flooded). This conservative assumption bounds any possible accident scenario, including the effects of a fuel handling accident or a misloading event. Additionally, the new fuel storage racks have the following design features which preclude flooding:

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19: EA-SFP-97-03 Table 4.8 Case *"nf41a"*.

20: EA-SFP-97-03 Table 4.9 Case *"nf50na"*.

1. All cells and spaces between cells have openings at the bottom to facilitate draining.

2. The rack is situated above a coarse steel grating floor. The floor below the grating is approximately another 12 ft.

## 4.0 REGION I FUEL STORAGE RACK 95/95 $k_{eff}$ CALCULATION

The results of the criticality analysis for the Palisades Region I fuel storage racks are presented in this section [21]. The Region I spent fuel storage rack design being evaluated is an existing array of fuel racks that was previously qualified for storage of various 15x15 fuel assembly types having maximum enrichments up to 4.40 w/o $^{235}U$ [22]. The Region I racks have two separate geometries. The "E-type" rack has a slightly higher center-to-center spacing than the "main pool" racks. The main pool rack geometry is more limiting [23]. Therefore the main pool rack is used for the Region I criticality evaluation presented here. Figure 5 shows the limiting Region I rack cell. The calculation presented in this report shows that the 95/95 $k_{eff}$ remains below 0.95 for enrichments of up to 4.95 w/o $^{235}U$ with no credit for the presence of boron in the spent fuel pool water.

## 4.1 METHODS

The CASMO computer code is used to perform a sensitivity study that quantifies, in terms of reactivity ($\Delta k$), the impact of possible variations in material characteristics and dimensions within manufacturing tolerances for both the fuel and the racks. The maximum $\Delta k$ for each tolerance is quantified and used in the determination of the 95/95 $k_{eff}$. These tolerance uncertainties are discussed in detail in Section 4.3, "*Region I Uncertainty Development*".

Next, the MONK code is run to determine the calculated $k_{eff}$ for nominal rack geometries, 4.95 wt% assembly enrichment, and 0 ppm boron in the fuel pool. The uncertainty due to manufacturing tolerances is then combined with the statistical uncertainty of the MONK cases using the Square Root Sum of the Squares (SRSS) method. The 95/95 $k_{eff}$ is calculated by adding the combined SRSS uncertainty term and a temperature bias to the MONK calculated $k_{eff}$ reference reactivity.

## 4.2 ASSUMPTIONS

In addition to the assumptions listed in Section 2.3, the following assumptions are used to determine the 95/95 $k_{eff}$ for fuel assemblies stored in the Palisades fuel pool Region I racks

21: Details of the calculation and analysis can be found in EA-SFP-97-02 and EA-SFP-97-01.

22: EMF-91-174(P), "*Criticality Safety Analysis for the Palisades Spent Fuel ...*"

23: EASFP-97-01 Section 4.7, "Results".

1. The fuel assembly array is infinite in lateral (x and y) extent and a 30 cm water reflector is modeled on the top and bottom of the fuel.

2. The Spent Fuel Pool moderator is water with a zero ppm concentration of soluble boron. A water density of 1.0 $gm/cm^3$ is used.

3. $^{10}B$ loading in the rack poison sheets is conservatively modeled as less than the minimum manufactures reported areal density of 0.0959 g/cc [24].

4. All storage cells are loaded with fresh 4.95 wt% enriched fuel assemblies.

## 4.3 REGION I UNCERTAINTY DEVELOPMENT

The reactivity effects related to variations within manufacturing tolerances of the fuel and Region I storage racks are presented in this section. These variations are quantified using CASMO-3 [25]. Both positive (+) and negative (-) tolerances are evaluated. The maximum positive reactivity effect is presented below. These reactivity effects are determined with 4.95 wt% fuel and no soluble boron in the pool water. They are combined with the MONK calculation uncertainty (2s) using the square root sum of the squares (SRSS) method yielding the total uncertainty. The statistical combination of uncertainties is also detailed in Table 2. Appropriate biases are also considered in the final 95/95 $k_{eff}$ determination.

**Fuel Pellet Density:** A(±) 1.5 % variation about a nominal percent theoretical density of 96.0% is considered. The resulting change in reactivity is $\Delta k = 0.00151$.

**Fuel Pellet Diameter:** A (±) 0.0005 inch variation about the nominal pellet diameter of 0.3600 inches is considered. The resulting change in reactivity is $\Delta k = 0.00024$.

**$^{235}U$ Enrichment:** The enrichment tolerance of (±) 0.05 w/o $^{235}U$ about the nominal reference enrichment of 4.95 w/o $^{235}U$ is considered. The resulting change in reactivity is $\Delta k =$ 0.00156.

**Cladding ID:** A (±) 0.0015 inch variation about the nominal cladding ID of 0.3670 inches is considered. The resulting change in reactivity is $\Delta k = 0.00006$.

**Cladding OD:** A (±) 0.002 inch variation about the nominal cladding OD of 0.4170 inches is considered. The resulting change in reactivity is $\Delta k = 0.00302$.

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24: Discussion found in EA-SFP-97-01 Section 4.2, *"Material Properties"*.

25: EA-SFP-97-01 Table 4.7.

**Storage Cell Pitch:** A (±) 0.04 inch tolerance about the nominal 10.25 inch reference cell pitch is considered. The resulting change in reactivity is $\Delta k = 0.00506$.

**$B_4C$ Panel Thickness:** A (±) 0.02 inch tolerance about the nominal 0.21 inch $B_4C$ panel thickness is considered. The resulting change in reactivity is $\Delta k = 0.00403$

**$B_4C$ Panel Width:** A (±) 0.02 inch tolerance about the nominal 8.26 inch $B_4C$ panel width is considered. The resulting change in reactivity is $\Delta k = 0.00125$.

**$^{10}B$ Areal Density:** A -10% tolerance about the nominal $^{10}B$ density is considered. The resulting change in reactivity is $\Delta k = 0.0005$.

**Can Inner Wall Thickness:** A (±) 0.01 inch tolerance about the can inner wall thickness of 0.125 inches is considered. The resulting change in reactivity is $\Delta k = 0.00033$.

**Can Outer Wall Thickness:** A (±) 0.01 inch tolerance about the can outer wall thickness of 0.125 inches is considered. The resulting change in reactivity is $\Delta k = 0.00213$.

**Can Outer Wall Thickness:** A (±) 0.01 inch tolerance about the can outer wall thickness of 0.125 inches is considered. The resulting change in reactivity is $\Delta k = 0.00073$.

**Calculation Uncertainty:** The statistical uncertainty in the MONK Monte Carlo calculations (2*s ≈ 95% confidence interval for a normal distribution) is considered. The resulting change in reactivity is $\Delta k = 0.0012$.

## BIASES

**Methodology Bias:** MONK consistently over predicts the $k_{eff}$ for the types of criticality evaluations being performed here. Any calculated methodology bias would be negative and its use would be less conservative than using the unbiased value. See Section 9.0, *"Computer Code Benchmarking"*, for a more detailed discussion.

**Water Temperature Bias:** A reactivity bias of 0.0012 is applied to account for the effect of the normal range of spent fuel pool water temperatures (40°F to 150°F).

## 4.4 RESULTS

MONK is used to calculate the nominal $k_{eff}$ with no credit for soluble boron. MONK is also used to determine the $k_{eff}$ for the worst case assembly position and the worst case shifting of the $B_4C$ plates. The results show that the offset poison case resulted in the highest nominal $k_{eff}$ of 0.9357. The 95/95 $k_{eff}$ for the Region I spent fuel rack configuration is developed by adding the temperature bias and the statistical sum of independent tolerance uncertainties to the calculated MONK $k_{eff}$. The summation is shown below and summarized in Table 2. The result is a 95/95 $k_{eff}$ of 0.9449 [26].

$$k_{eff_{95/95-No\ Boron}} = k_{MONK-NB} + \Delta k_{\Delta T} + \sqrt{(2s_{MONK-NB})^2 + s_{tol}^2}$$

Where:

| | |
|---|---|
| $k_{eff95/95}$ | is the 95/95 $k_{eff}$ of 4.95 wt% fresh fuel with no Boron. |
| $k_{MONK-NB}$ | is the MONK calculated $k_{eff}$ for 4.95 wt% fresh fuel with no Boron. |
| $2s_{MONK-NB}$ | is 2x the MONK calculation standard deviation (No Boron) = .0012 Δk |
| $s_{Tol}$ | is the 95/95 reactivity effect of manufacturing tolerances = 0.00792 Δk. |
| $\Delta k_{\Delta T}$ | is the reactivity effect of variations in temperature = 0.0012 Δk. |

Under normal conditions, the $k_{eff}$ of the Region I racks, including all mechanical and calculational uncertainties, is below 0.95 with a 95 percent probability at a 95 % confidence level with 0.0 ppm of natural boron in the fuel pool water. This section only addresses normal conditions. The analysis of accident conditions such as a dropped assembly, a misloaded assembly, and the interaction between Region I racks and Region II racks is discussed in Section 7.0, *"Rack Interaction, Abnormal Conditions and Postulated Accidents"*.

## 5.0 REGION II FUEL STORAGE RACK 95/95 $k_{eff}$ CALCULATION

The results of the criticality analysis for the Palisades Region II spent fuel storage racks are presented in this section [27]. The Region II spent fuel storage rack design being evaluated is an existing array of fuel racks containing the neutron absorber Boraflex. This design was previously qualified for storage of 15x15 pin fuel assemblies, with enrichments up to 3.27 w/o $^{235}U$, meeting the applicable initial enrichment-burnup requirements [28]. The calculation presented here takes no credit for the presence of Boraflex in the racks. The omission of Boraflex from the calculation is compensated for by taking credit for the presence of soluble boron in the spent fuel pool water and, an increase in the required assembly burnup for a given enrichment. The calculations show that $k_{eff}$ is less than 1.0 for the combinations of initial enrichment, burnup and decay time shown in Figure 9 with no credit for boron. The presence of 850 ppm ensures that the 95/95 $k_{eff}$ is less than 0.95 under normal storage conditions.

26: EA-SFP-98-02 Table 4.3, *"Offset Poison"*.

27: Details of the calculation and analysis can be found in EA-SFP-98-03 and EA-SFP-98-04.

28: WNEP-8626, *"Design Report of Region Two Spent Fuel Storage Racks:...."*.

## 5.1 METHODS

The CASMO computer code is used to calculate a $k_\infty$ for a fresh fuel assembly with an enrichment of 1.14 wt% and no boron present in the pool. This calculated $k_\infty$ is referred to as $k_{ref}$. The 1.14 wt% enrichment is chosen because it will ensure that the final $k_{eff}$ calculated by the MONK computer code will remain below 1.0 for the un-borated condition and below 0.95 when boron is credited.

The CASMO computer code is also used to perform a sensitivity study that quantifies, in terms of reactivity ($\Delta k$), the impact of possible variations in material characteristics and dimensions within manufacturing tolerances for both the fuel and the racks. The maximum $\Delta k$ for each tolerance is quantified and used in the determination of the 95/95 $k_{eff}$. These tolerance uncertainties are discussed in detail in Section 5.3, "*Region II Uncertainty Development*".

Next the MONK code is run to determine the calculated $k_{eff}$ for nominal rack geometries, 1.14 wt% assembly enrichment, and 0 ppm boron in the fuel pool. The uncertainty due to manufacturing tolerances is then combined with the statistical uncertainty of the MONK case using the Square Root Sum of the Squares (SRSS) method. The 95/95 $k_{eff\text{-}no\ boron}$ is calculated by adding the combined SRSS uncertainty term and a temperature bias to the MONK calculated $k_{eff}$.

CASMO is also used to apply the concept of reactivity equivalencing. Reactivity equivalencing is predicated upon the reactivity decrease associated with fuel depletion (burnup). Quantifying this reactivity decrease as a function of burnup vs. assembly enrichment allows the storage of fuel assemblies with enrichments greater than 1.14 wt% in the Region II racks. This is accomplished by running a series of CASMO calculations. These calculations use $k_{ref}$ as a target value while finding various burnup and enrichment combinations that yield $k_{eff}$ less than or equal to $k_{ref}$. The end result is a plot of enrichment vs. burnup ordered pairs. Assemblies with initial enrichment and burnup combinations that fall below the curve are not acceptable for storage in Region II. Assemblies with initial enrichment and burnup characteristics equal to or above the curve are acceptable for storage in Region II.

The use of the 2D CASMO-3 code to perform reactivity equivalencing ignores the effects of the burnup profile axially along the assembly. This is of note because assembly ends which are burned significantly less than the assembly average can have considerable effects on reactivity. Therefore, Consumers Energy evaluated the axial burnup profiles of discharged assemblies in the Palisades spent fuel pool in order to quantify any possible positive reactivity effects. The result of the evaluation is a set of adjustment factors that are used to bias the target $k_{ref}$ in the reactivity equivalencing calculations discussed earlier. These adjustment factors ranged from a $\Delta k$ of 0.00337 at an enrichment of 2.00 wt% $^{235}U$ to a $\Delta k$ of 0.01542 at an enrichment of 4.6 wt% $^{235}U$ [29]. The use of these adjustment factors ensures that reactivity effects due to assembly ends which are burned significantly less than the assembly average burnup, are conservatively accounted for.

Figure 9 shows a family of curves that represent combinations of fuel enrichment and discharge burnup which yield the same rack multiplication factor ($k_{eff}$) as the rack loaded with 1.14 w/o $^{235}U$ fuel (at zero

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29: Analysis documented in EA-CRIT-99-01.

burnup). Each curve represents a different decay time. Decay Time credit is an extension of the burnup credit process. Decay time credit takes into account the time an assembly has been discharged. The radioactive decay of isotope to daughter isotopes over the time that the assembly has been discharged results in reduced reactivity. This reduction in assembly reactivity translates into a reduction in the minimum burnup required at a given initial enrichment.

Finally, the MONK code is run again to determine the calculated $k_{eff}$ for nominal rack geometries, 1.14 wt% assembly enrichment, and 850 ppm of boron in the fuel pool. The 95/95 $k_{eff\text{ - }850\text{ boron}}$ is then developed by adding to the MONK calculated $k_{eff}$ the combined SRSS uncertainty term, a temperature bias, a bias to account for reactivity equivalencing methodology uncertainties, a bias to account for uncertainties in plant assembly burnup records and, a bias to account for the change in boron worth with assembly burnup.

## 5.2 ASSUMPTIONS

In addition to the Assumptions listed in Section 2.3 the following assumptions are used to determine the 95/95 $k_{eff}$ for fuel assemblies stored in the Palisades spent fuel pool Region II racks:

1. The fuel assembly array is infinite in lateral (x and y) extent and a 30 cm water reflector is modeled on the top and bottom of the fuel.

2. All storage cells are loaded with fuel assemblies.

3. The reactivity calculation for exposed fuel ignores Xe.

4. No credit is taken for the presence of Boraflex poison panels in the Region II fuel storage racks. The Boraflex volume is replaced with pure (no boron) water.

5. The following conservative core conditions are assumed for the assembly depletion calculation.
   a. A reactor coolant soluble boron concentration of 700 ppm .
   b. A reactor coolant temperature of 580.2 °K
   c. A fuel temperature of 800 °K
   d. A power density of 50.0 W/gU
   e. A power history of one continuous cycle.

6. Conservative reactivity adjustments based on limiting axial burnup profiles are assumed for each burnup/enrichment point on the burnup credit curve.

## 5.3 REGION II UNCERTAINTY DEVELOPMENT

The reactivity effects related to tolerance and material variations associated with the manufacture of the fuel and Region II storage racks are presented in this section. These variations are quantified using CASMO-3 [30]. Both positive (+) and negative (-) tolerances are evaluated. The maximum positive reactivity effect is presented below. The reactivity effects are determined with 1.14 wt% fuel and no soluble boron in the pool water. They are combined using the square root sum of the squares (SRSS) method yielding the total uncertainty. The statistical combination of uncertainties is also detailed in Table 3. Appropriate biases are also considered in the final 95/95 $k_{eff}$ determination.

**Fuel Pellet Density:** A (±) 1.5 % variation about a nominal reference percent theoretical density of 96.0% is considered. The resulting change in reactivity is $\Delta k = 0.00215$.

**Fuel Pellet Diameter:** A (±) 0.0005 inch variation about the nominal pellet diameter of 0.3600 inches is considered. The resulting change in reactivity is $\Delta k = 0.00033$.

**$^{235}$U Enrichment:** A (±) 0.05 w/o $^{235}$U variation about the nominal reference enrichment of 1.14 w/o $^{235}$U is considered. The resulting change in reactivity is $\Delta k = 0.01588$.

**Cladding ID:** A (±) 0.0015 inch variation about the nominal cladding ID of 0.3670 inches is considered. The resulting change in reactivity is $\Delta k = 0.00039$.

**Cladding OD:** A (±) 0.002 inch variation about the nominal cladding OD of 0.4170 inches is considered. The resulting change in reactivity is $\Delta k = 0.00054$.

**Storage Cell ID:** A (±) 0.025 inch tolerance about the nominal storage cell ID of 9.0 inches is considered. The resulting change in reactivity is $\Delta k = 0.00268$.

**Storage Cell Pitch:** A (±) 0.06 inch tolerance about the nominal cell pitch of 9.17 inches is considered. The resulting change in reactivity is $\Delta k = 0.00627$.

**Stainless Steel Thickness:** A (±) 0.007 inch tolerance about the nominal rack structure stainless steel thickness of 0.095 inches (wall + wrapper) is considered. The resulting change in reactivity is $\Delta k = 0.00773$.

**Calculation Uncertainty:** The statistical uncertainty in the MONK Monte Carlo calculations (2*s ≈ 95% confidence interval for a normal distribution) is considered. The resulting change in reactivity is $\Delta k = 0.0006$.

30: EA-SFP-98-03 Table 4.10.

## BIASES

**Methodology Bias:** MONK consistently over predicts the $k_{eff}$ for the types of criticality evaluations being performed here. Any calculated methodology bias would be negative and its use would be less conservative than using the unbiased value. See Section 9, *"Computer Code Benchmarking"*, for a more detailed discussion.

**Water Temperature Bias:** A reactivity bias of 0.00004Δk is applied to account for the effect of the normal range of spent fuel pool water temperatures (40°F to 150°F).

**Soluble Boron Credit Bias:** A reactivity bias of 0.07382 Δk is applied to account for the reduction in the effectiveness of boron when considered with burned fuel.

**Plant Exposure Records Bias:** A reactivity bias of 0.03349 Δk is applied to account for the uncertainty associated with plant exposure records.

**Reactivity Equivalencing Method Bias:** A reactivity bias of 0.025 Δk is applied to account for the method uncertainty associated with the reactivity equivalencing calculation.

## 5.4 RESULTS

The MONK calculation of the Region II nominal $k_{eff}$ with no credit for soluble boron in the spent fuel pool yields a $k_{eff}$ of 0.9796. The 95/95 $k_{eff}$ for the Region II spent fuel rack configuration is developed by adding the temperature bias and the statistical sum of independent tolerance uncertainties to the nominal MONK reference reactivity. The summation is shown below and results in a 95/95 $k_{eff}$ of 0.9987 [31].

$$k_{eff_{95/95\text{-No Boron}}} = k_{MONK\text{-}NB} + \Delta k_{\Delta T} + \sqrt{(2s_{MONK\text{-}NB})^2 + s_{tol}^2}$$

Where:

| | |
|---|---|
| $k_{95/95\text{-No Boron}}$ | is the 95/95 $k_{eff}$ of 1.14 wt% fresh fuel equivalent with no Boron. |
| $k_{MONK\text{-}NB}$ | is the MONK calculated $k_{eff}$ for 1.14 wt% fresh fuel with no Boron. |
| $2s_{MONK\text{-}NB}$ | is 2x the MONK calculation standard deviation (No Boron) = 0.0006 Δk |
| $s_{Tol}$ | is the 95/95 reactivity effect of manufacturing tolerances = 0.01907 Δk. |
| $\Delta k_{\Delta T}$ | is the reactivity effect of variations in the pool temperature = 0.00004 Δk. |

This shows that under normal storage conditions the Palisades spent fuel racks will remain subcritical ($k_{eff}$ < 1.0), when loaded with 1.14 w/o $^{235}U$ fuel assemblies and no soluble boron is present in the spent fuel pool water. Next, the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ including all tolerances and

31: EA-SFP-98-04, Table 4.7, case *"reg23d"*.

uncertainties is determined. This soluble boron credit is used to show the additional safety margin due to the presence of boron in the spent fuel pool water.

**Soluble Boron Credit $k_{eff}$ Calculation**

The MONK calculation of $k_{eff}$ with 850 ppm of soluble boron in the spent fuel pool results in a $k_{eff}$ of 0.7964. The 95/95 $k_{eff}$ for the Region II spent fuel rack configuration with credit for boron is developed by adding the same temperature bias and statistical uncertainties discussed in the no boron case to the nominal MONK reference reactivity. In addition, biases related to reactivity equivalencing, boron credit methods, and uncertainty in plant exposure records are included when crediting boron. The summation is shown below and results in a 95/95 $k_{eff}$ of 0.9478 [32].

$$k_{eff_{95/95\text{-}850}} = k_{MONK\text{-}850} + \Delta k_{\Delta T} + \Delta k_B + \Delta k_{RE} + \Delta k_R + \sqrt{(2s_{MONK\text{-}850})^2 + s_{tol}^2}$$

Where:

| | |
|---|---|
| $k_{eff95/95\text{-}850}$ | is the 95/95 $k_{eff}$ of 1.14 wt% fresh fuel equivalent with 850 ppm Boron. |
| $k_{MONK\text{-}850}$ | is the MONK calculated $k_{eff}$ for 1.14 wt% fresh fuel with 850 ppm of Boron. |
| $2s_{MONK\text{-}850}$ | is 2x the MONK calculation standard deviation (850 ppm) = 0.0006 $\Delta$k. |
| $s_{Tol}$ | is the 95/95 reactivity effect of manufacturing tolerances = 0.01907 $\Delta$k. |
| $\Delta k_T$ | is the reactivity effect of variations in temperature = 0.00004 $\Delta k_T$. |
| $\Delta k_B$ | is the lowered reactivity effect of boron with burned fuel = 0.07382 $\Delta$k. |
| $\Delta k_{RE}$ | is the bias to offset reactivity equivalencing method uncertainty = 0.025 $\Delta$k. |
| $\Delta k_R$ | is the bias to offset uncertainty in plant exposure records = 0.03349 $\Delta$k. |

The 850 ppm boron is chosen to yield a $k_{eff}$ which is conservatively low so that burnup calculation and measurement uncertainties are bounded. The CASMO-3 burnup uncertainty is conservatively approximated by taking 5.0% of the largest reactivity defect attributed to burnup, $\Delta k_{RE}$ = 0.05*0.42733 = 0.02137. This uncertainty is then rounded up to 0.025 $\Delta$k.

The uncertainty in reactivity equivalencing due to axial burnup shape effects is conservatively applied to the target $k_{eff}$ ($k_{ref}$) utilized in the reactivity equivalencing calculation as discussed earlier. A further uncertainty is added, $\Delta k_R$, to account for a 10% variation in plant assembly burnup records.

The worth of a ppm of boron is reduced as fuel burnup is increased. To account for this reduction, $\Delta k_B$ is added to the $k_{eff95/95\text{-}850}$ to ensure that the calculation with 1.14 wt% fresh fuel bounds equivalent enrichment/burnup combinations.

The curves presented in Figure 9 are used to determine the eligibility of assemblies for storage in the Region II racks. They are developed to ensure that $k_{95/95\text{-No Boron}}$ remains below 1.0 (subcritical). Assemblies are classified by their maximum average planar initial enrichment in wt% $^{235}$U. If an assembly has an average

32: EA-SFP-98-04, Table 4.7, Case *"850breg2"*.

burnup which falls above the "no decay" line then that assembly is acceptable for storage immediately following discharge from the reactor. If the assembly average burnup falls between the "no decay" line and the "1 yr decay" line then the assembly must cool for one year in Region I before being moved into Region II. If burnup falls between the "1 yr decay" line and the "3 yr decay" line then the assembly must cool 3 years etc .... If the assembly average burnup is below the "8 yr decay" line the assembly cannot be stored in Region II without further analysis. Any assembly with a maximum average planar enrichment below 1.14 can be stored without any burnup. Assemblies between 1.14 and 1.2 wt% require 3.477 GWD/MTU for storage.

Table 4 provides the data used to create Figure 9. It is acceptable to linearly interpolate between points to determine the required burnup for assemblies with initial enrichments not explicitly stated [33]. In addition, consideration of a 10% uncertainty in assembly burnup records is accounted for in the development of $k_{95/95\text{-}850ppm}$. It is therefore, NOT necessary to reduce stated burnup by a measurement uncertainty up to 10% when evaluating an assembly against the curves.

## 6.0 FUEL ELEVATOR/FUEL TRANSFER MACHINE 95/95 $k_{eff}$ CALCULATION

The results of the criticality analysis for the Palisades fuel elevator [34] and fuel transfer machine [35] are presented in this section. The fuel transfer machine is used to transfer fuel from the spent fuel pool to the reactor cavity through the transfer canal. The transfer machine is capable of holding two assemblies. Although Palisades typically moves only one assembly through the transfer canal at a time there are other scenarios that would result in two assemblies being present in the transfer machine simultaneously. In fact, it is common during a core shuffle to place a second assembly into the fuel transfer machine before unloading the first assembly.

The fuel elevator/inspection station is also capable of holding two assemblies and is used to perform multiple tasks. Primarily it is used to transfer un-irradiated fuel from the new fuel storage rack into the spent fuel pool for transport to the reactor via the fuel transfer machine. The elevator is also used to inspect and reconstitute irradiated fuel. Both of these components were previously qualified for holding two 15x15 pin Palisades fuel assemblies with a maximum planar average enrichment up to 4.40 w/o $^{235}U$ and a minimum pool boron concentration of 600 ppm. Figure 7 and Figure 8 show the fuel elevator and fuel transfer machine geometry respectively.

The calculations presented here show that the presence of 850 ppm of boron in the pool water ensures that the 95/95 $k_{eff}$ remains below 0.95 for two fresh 4.95 wt% assemblies in the transfer machine. The 850 ppm is not required for normal operations in the elevator/inspection station (i.e 95/95 $k_{eff}$ is below 0.95 with 0.0 ppm boron). However, the unlikely close approach of an assembly in the fuel handling machine mast with an assembly in a raised elevator is conservatively shown to result in a 95/95 $k_{eff}$ less than 0.95 when 850 ppm

33: EA-SFP-98-04, Section 5.4, "*Burnup Curves*".

34: Details of the calculation and analysis can be found in EA-SFP-98-02.

35: Details of the calculation and analysis can be found in EA-SFP-98-01.

boron is present in the pool water. The basis for crediting boron when analyzing these fuel handling scenarios is discussed further in Section 7.4, "*Required Boron*", and Section 7.5, "*Boron Credit Discussion*".

## 6.1 METHODS

The MONK code is run to determine the 95/95 $k_{eff}$ for the fuel elevator. The MONK calculations took into account the impact of manufacturing tolerance variations on the fuel dimensions and assembly placement in the fuel elevator as well as any possible interaction with a third assembly. Since the "worst case" manufacturing tolerances and assembly loadings are considered in the MONK calculation, the determination of the 95/95 $k_{eff}$ from the MONK calculated $k_{eff}$ is relatively straight forward. The calculated $k_{eff}$ is added to two times the standard deviation in the MONK calculational results to determine the 95/95 $k_{eff}$.

Similarly, the MONK code is run to determine the 95/95 $k_{eff}$ for the fuel transfer machine. Two fresh assemblies are considered at the minimum separation allowed by the transfer machine structural material. The worst case manufacturing tolerances are considered in the MONK calculation making the determination of the 95/95 $k_{eff}$ from the MONK calculated $k_{eff}$ straight forward.

## 6.2 ASSUMPTIONS

The following assumptions are used to determine the 95/95 $k_{eff}$ for fuel assemblies stored in the Palisades fuel elevator:

1. The smallest possible separation between two assemblies in the fuel elevator or the fuel transfer machine (worst case center-to-center spacing) is conservatively assumed.

2. The structural material of the elevator and the transfer machine is ignored.

3. The Region I fuel racks are assumed to be separated from the elevator by a distance which is conservatively small.

4. The close approach of two assemblies is conservatively modeled as two fresh 4.95 wt% assemblies in contact with each other in the elevator geometry.

## 6.3 GEOMETRY SPECIFICATIONS

The MONK analysis sets key parameters that describe the fuel assembly and fuel handling component geometry to conservative values within their expected manufacturing tolerances. In general, the cladding is made as thin as possible while the pellet is made as large and as dense as possible. The combination of parameters used to develop the MONK model are presented below and conservatively represent the highest reactivity assembly possible within current manufacturing tolerances.

**Fuel Pellet Density:** A (+) 1.5 % variation about a nominal percent theoretical density of 96.0% is modeled.

**Fuel Pellet Diameter:** A (+) 0.0005 inch variation about the nominal pellet diameter of 0.3600 inches is modeled.

**$^{235}U$ Enrichment:** The enrichment tolerance of (+) 0.05 w/o $^{235}U$ about the nominal reference enrichments of 4.95 w/o $^{235}U$ is modeled.

**Cladding ID:** A (+) 0.0015 inch variation about the nominal cladding ID of 0.3670 inches is modeled.

**Cladding OD:** A (-) 0.002 inch variation about the nominal cladding OD of 0.4170 inches is modeled.

**North-South Offset (Elevator):** The minimum possible centerline North-South offset of 1.0203 inches is modeled.

**East-West Separation (Elevator):** The worst case East-West separation of two assemblies in the fuel elevator of 7.945 inches is modeled.

**Assembly Separation (Transfer Machine):** The worst case separation of two assemblies in the fuel transfer machine of 0.5 inches is modeled.

## 6.4 RESULTS

The 95/95 $k_{eff}$ is developed by adding two times the MONK calculation standard deviation to the MONK calculated value as demonstrated by the equation below. Under normal conditions (ie, no close approach of two assemblies) the elevator $k_{eff}$ is determined to be 0.9312 at a 95% probability with 95% confidence assuming 0.0 ppm of boron in the pool water. The fuel transfer machine 95/95 $k_{eff}$ assuming 850 ppm of boron is determined to be 0.9453. The most reactive scenario occurs when two fresh 4.95 wt% enriched fuel assemblies are modeled side by side in the fuel elevator. The calculated $k_{eff}$ for that case assuming a fuel pool boron concentration of 850 ppm is 0.9465. Incorporating the MONK standard deviation (2s=0.0014), the calculated 95/95 $k_{eff}$ for this configuration is 0.9479.

$$k_{eff_{95/95}} = k_{MONK} + 2s_{MONK}$$

Where:

| | |
|---|---|
| $k_{eff95/95}$ | is the 95/95 $k_{eff}$ of 4.95 wt% fresh fuel. |
| $k_{MONK}$ | is the MONK calculated $k_{eff}$. |
| $2s_{MONK}$ | is 2x the MONK calculation standard deviation. |

The removal of fuel pins from the assembly during fuel inspection and/or assembly reconstitution activities is also evaluated. The calculations show that the removal of pins in the presence of borated water results in a reduction of $k_{eff}$. The results demonstrate that when a minimum boron concentration of 850 ppm is maintained in the fuel pool water, $k_{eff}$ is below 0.95 at a 95% probability with a 95% confidence for all postulated configurations of 4.95 wt% fuel assemblies loaded into the fuel elevator or fuel transfer machine.

## 7.0 RACK INTERACTION, ABNORMAL CONDITIONS, POSTULATED ACCIDENTS

The previous sections have shown that the 95/95 $k_{eff}$ for normal conditions will remain below 0.95 as long as the fuel pool boron concentration is above 850 ppm. This section describes the evaluation of any possible interactions between storage racks and/or fuel handling equipment. It also presents the postulated accident conditions and the associated increase in reactivity due to each accident [36]. The boron concentration that will offset the largest reactivity increase from the accident situations is determined. Finally, the section addresses possible events which might cause a dilution (reduction in boron concentration) of the Palisades fuel pool.

## 7.1 RACK INTERACTION

The interaction between the Region I rack and an assembly in the elevator or fuel handling machine are bounded by the conservative calculation of the elevator 95/95 $k_{eff}$ presented in Section 6.0 (Calculations include surrounding Region I storage cells).

The interaction of two Region I rack modules and between two Region II modules is bounded by the conservative modeling approach used which assumes that the rack is a single structure with no spacing in-between modules. Region I racks contain $B_4C$ poison material around all rack walls. Region II racks do not have Boraflex on the outer wall of the rack module. This lack of poison between rack modules is accounted for since Boraflex is not considered when determining the 95/95 $k_{eff}$.

The interaction between Region I and Region II racks is evaluated assuming the minimum allowed 2 inch separation between racks. The evaluation shows that $k_{eff}$ calculated considering the interface between regions is conservatively bounded by the 95/95 $k_{eff}$ determined for Region I and Region II individually [37].

## 7.2 POSTULATED ACCIDENTS

Evaluation of accident conditions in the Palisades fuel pool take no credit for Boraflex in the Region II racks. Conservative assumptions for fuel design and boron composition reported in Section 2.0 are applied. The Double Contingency principle of ANSI ANS 8.1-1983 is applied. This states that one is not required to

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36: Details of the calculation and analysis can be found in EA-SFP-99-02.

37: EA-SFP-99-02, Section 4.8.1, "*Rack Interaction Results*".

assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional (amounts above 850 ppm required for normal Region II storage) soluble boron in the pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. The postulated accident scenarios addressed include:

1. Optimum Moderation Accident
2. Loss of fuel pool cooling resulting in an increase in fuel pool temperature
3. Misplacement of a fresh fuel assembly into Region II, burn-up credit racks
4. Fuel assembly drop outside of a Region II rack module (cask loading area)
5. Fuel assembly drop on top of a Region I or Region II rack

### 7.2.1 Optimum Moderation Accident

The determination of the 95/95 $k_{eff}$ presented in Sections 3.0 through 5.0 for the new fuel storage racks, Region I and Region II considered the optimum moderation scenario. The fuel pool is normally fully flooded and variations in water density are only credible during rack installation and removal. The new fuel storage rack is normally dry with a $k_{eff} < 0.6$. It can be postulated that an event such as the use of a fire hose might cause a mist in the rack area leading to an optimum moderation conditions. The new fuel storage rack 95/95 $k_{eff}$ evaluation is performed at the optimum moderation condition which is shown to be fully flooded (1g/cc). The successful evaluation under these conditions bounds any possible accident scenario, including the effects of a fuel handling accident or a misloading event in the new fuel storage rack. The following design features of the new fuel storage racks preclude flooding:

1. All cells and spaces between cells have openings at the bottom to facilitate draining.

2. The rack is situated above a coarse steel grating floor. The floor below the grating is approximately another 12 ft.

### 7.2.2 Loss of Fuel Pool Cooling Accident

The loss of spent fuel pool cooling has an adverse effect on calculated $k_{eff}$. CASMO-3 calculations are performed to determine the increase in $k_{eff}$ following a loss of fuel pool cooling. The Region I and Region II 95/95 $k_{eff}$ evaluation under normal conditions (Sections 3.0 and 4.0) considers temperatures up to 150 °F. The calculation for the loss of fuel pool cooling assumes the pool temperature reachs 240 °F (conservatively above boiling). This results in an increase in $k_{eff}$ of 0.0019 $\Delta k$ [38]. Loss of fuel pool cooling does not impact the new fuel storage rack because it is not located in the pool itself.

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38: EA-SFP-99-02, Section 4.8.2, "*Loss of Cooling Accident*".

### 7.2.3 Misloaded Assembly

The placement of a fresh 4.95 wt% assembly in Region II has an adverse effect on calculated $k_{eff}$. MONK calculations are performed to determine the increase in $k_{eff}$ due to this postulated misloading event. The case resulted in an increase in $k_{eff}$ of 0.0801 ± 0.0016 (2s) Δk under zero boron assumptions [39]. Misloading of the New Fuel Storage rack is bounded by the optimum moderation accident.

### 7.2.4 Assembly Dropped Between Rack Modules

The design of the new fuel storage rack and the Region I storage racks is such that it precludes the placement of an assembly between the racks or between the racks and the pool walls. The placement of an assembly beside the Region II racks when one of the 11x11 rack modules is removed for dry fuel storage canister loading is possible and does have an adverse effect on the calculated $k_{eff}$. MONK calculations show the resulting increase in $k_{eff}$ is 0.0354 ± 0.0014 (2s) Δk under zero boron assumptions [40].

### 7.2.5 Fuel assembly drop on top of rack

Both the Region I and Region II racks are designed to absorb the impact of a dropped assembly without experiencing significant deformation [41] [42]. The assembly sitting on the top of the racks will be separated from the active fuel column of the assemblies in storage by greater than 10 inches of borated water. The separation is adequate to preclude neutron interaction. Therefore, the reactivity increases due to the other accident conditions discussed here will bound the dropped assembly scenario for the Region I and Region II racks. Since the new fuel storage rack is normally dry with $k_{eff} < 0.6$, the optimum moderation accident is used to bound the drop of an assembly onto the new fuel storage rack.

## 7.3 OTHER ACCIDENTS / ABNORMAL CONDITIONS

Other accident and abnormal conditions which might result in the increase in reactivity can be postulated. The Palisades fuel pool area is not contained within a missile barrier. Therefore, it is plausible that a missile could be projected into the pool (during a tornado for example) and cause damage to the assemblies being stored. Any number of geometry configurations are possible. In general, presence of the minimum 1720 ppm of boron is relied upon to mitigate the effects of such an accident. Section 6 of this report addresses the bounding and unlikely occurrence of two assemblies coming into contact during fuel handling around the fuel elevator or transfer machine. It is shown that 850 ppm of boron is required to mitigate the effects of such a handling incident. Again, the presence of the minimum Technical Specification 1720 ppm is relied upon to mitigate the increase in $k_{eff}$.

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39: EA-SFP-99-02, Section 4.8.3, "*Misloaded Assembly*".

40: EA-SFP-99-02, Section 4.8.4, "*Assembly Dropped Beside Region II Rack Module*".

41: WNEP-8626, "*Design Report of Region Two Spent Fuel Storage Racks...*"

42: Docket No. 50-255, "*Spent Fuel Pool Modification Description and Safety Analysis*"

## 7.4 REQUIRED BORON

This evaluation has shown that a boron concentration of 850 ppm will ensure that the 95/95 $k_{eff}$ remains below 0.95 in the Region I and Region II racks under normal loading conditions and during normal fuel handling conditions in the fuel elevator and transfer machine. The limiting accident condition was determined to be the Misloaded Assembly event described in Section 7.2.3. An additional 500 ppm of soluble boron will mitigate the effect of this limiting rack misloading event [43]. Therefore, the overall requirement for spent fuel boron concentration to ensure under all normal and credible accident scenarios at a 95 percent probability and with a 95% confidence that $k_{eff}$ is below 0.95 is 1350 ppm (850 ppm + 500 ppm).

## 7.5 BORON DILUTION EVENT

As allowed by the *"double contingency principle"* [44], the presence of boron is assumed when evaluating fuel handling accidents and other unusual events. However, credit for boron during normal storage suggests the need for a more detailed investigation into boron dilution events. The possible sources and circumstances which could lead to the dilution of the Palisades spent fuel pool are analyzed [45]. Eleven events are identified and classified into two categories. Category 1(*"Direct Dilution"*) events are considered in order to quantify the response time required to identify and terminate dilutions before the pool reaches the 850 ppm required to maintain the 95/95 Region II $k_{eff}$ below 0.95. The starting point for the dilution analyses is conservatively chosen to be the Technical Specification minimum 1720 ppm. Of the seven category 1 events considered, the limiting (shortest) time available for operators to terminate the dilution is 9.8 hours [46]. All category 1 events would require 107,600 gallons of pool water to overflow the pool unnoticed before the 850 ppm limit is reached. Operators would have to ignore wet auxiliary building floors, rising dirty waste tank levels (floor drains empty into these tanks) and rising safeguards room sump levels.

Category 2 (*"Emergency Pool Refill"*) events are analyzed to identify scenarios where the Palisades spent fuel pool might be drained to the point of requiring emergency filling to restore pool cooling. Palisades responses to category 2 events are governed by procedure. The analysis identifies constraints on unborated

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43: EA-SFP-99-02, Section 4.8.5, "*Required Boron*".

44: ANSI/ANS 57.2 Section 6.2.1.4 states that "*criticality analysis shall demonstrate the criticality could not occur without at least two unlikely, independent and concurrent incidents or abnormal occurrences*". Since boron is normally present in the fuel pool, assumption of 0.0 ppm under accident conditions would represent a second, independent unlikely event. This is in agreement with guidance given in ANSI/ANS 8.1, Section 4.2.2, "*Double Contingency Principle*".

45: Details can be found in EA-WJB-00-01, "*Spent Fuel Pool Dilution Analysis*".

46: EA-WJB-00-01 identifies a 8 hour minimum time for the *"Miss-application of a portable Tri-Nuc filter system"* event. However, the analysis also shows that the only available source for the volume of water needed in this event is tank T-90 which can deliver water at a much slower rate (≈ 50 Hours).

water addition necessary in responding to such scenarios. While plant procedures will need to be revised to incorporate the results of the dilution analysis, operators are able to recover from each identified event without decreasing pool boron concentration below assumed levels.

If a dilution event was allowed to progress long enough to dilute the pool to less than 850 ppm boron (0.0 ppm assumed), the new fuel storage, Region I, and fuel elevator analysis show that the 95/95 $k_{eff}$ remains below 0.95 under normal storage and handling conditions. The Region II analysis shows $k_{eff}$ will be below 1.0 at 0.0 ppm boron. Hence, criticality will not occur even in the event of an undetected dilution. Since it is unreasonable to assume that operators ignore the many physical indicators of a dilution (sump alarms, wet floors etc ..) and since dilution to 0.0 ppm will result in $k_{eff} < 1.0$, a dilution to critical during normal storage in the Palisades spent fuel pool is not a credible event.

Furthermore, 850 ppm of boron is required to ensure a 95/95 $k_{eff} < 0.95$ for the Palisades transfer (tilt) machine [47]. The 95/95 $k_{eff}$ of the tilt machine fully loaded with two 4.95 wt% enriched assemblies will be greater than 1.0 if 0.0 ppm boron is assumed. Additionally, use of the tilt machine would mean that the fuel pool water volume would be connected to the refueling cavity via the north tilt pit. It also would mean that boron concentrations are at refueling boron levels (typically > 2000 ppm) [48]. These conditions require a dilution to go unnoticed for much longer than reported here for category 1 events since the volume to be diluted is larger and starts at a higher boron concentration. Furthermore, dilutions in these conditions are bounded by Final Safety Analysis Chapter 14 dilution analyses [49]. Criticality concerns in the core (≈ 1300 ppm) occur long before approaching the 850 ppm required to maintain $k_{eff}$ below 0.95 in the tilt machine. In addition, Palisades administrative procedures prohibit activities which could lead to significant dilutions during fuel handling activities [50]. Prohibition of dilution activities and weekly boron sampling frequency [51] provide an added level of assurance that the 1350 ppm assumed under accident conditions is present.

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47: EA-SFP-98-01, "*Palisades Transfer Machine Criticality Calculations*".

48: SOP-28, Section 6.0, "*Initial Conditions*".

49: FSAR, Section 14.3, "*Boron Dilution*".

50: SOP-27 Attachment 2, "*Addition of Water to the Spent Fuel Pool or Refueling Cavity*".

51: **Docket No. 50-255 License DPR-20**, "*Improved Technical Specifications*", Section 3.7.15 implies 7 day surveillance interval when fuel is being moved. Section 3.9.1 requires a 72 hour sampling frequency in Mode 6, typical plant condition for fuel movement.

## 8.0 ANALYSIS CONSERVATISM

The criticality calculations presented by this report did not take credit for the conservative calculation bias introduced by the use of the MONK computer code. The criticality calculations incorporated many other conservative assumptions. Some of the more significant assumptions regarding the modeling of the fuel assembly, storage array, and the burnup credit calculation are presented below.

Fuel Assembly

1. An average planar enrichment is used for all pin locations rather than using designed pin enrichment distributions.
2. Fuel pellet dishing is ignored.
3. Burnable poisons in the fuel are ignored.

Storage Array

1. Boraflex Poison in the Region II racks is ignored. The reactivity effect of the including the Boraflex results in an approximate reduction in $k_{eff}$ of 0.1465 $\Delta k$.
2. $k_{eff}$ is determined assuming a full rack of fresh assemblies with no burnable poisons.
3. The $^{10}B$ poison loading of the Region I racks are conservatively calculated.
4. The normally dry new fuel storage rack is modeled flooded with pure water at a density giving optimum moderation (1 g/cc)
5. The fuel pool water temperature is modeled at 40 °F, much lower than typical spent fuel pool temperatures.
6. Spent fuel pool water is assumed to be un-borated for the new fuel and Region I rack calculations.
7. When considered, soluble boron in the pool water is modeled as only 17% $^{10}B$ atoms.

Burn-up vs. Enrichment Curve Development

1. No credit is taken for stored assemblies which have burnup and cooling times greater than required for their initial enrichment.

2. Boron concentration requirements are determined considering minimum required burnup on the maximum initial enrichment of 4.60 wt%. This artificially lowers the boron worth compared to what would be calculated at a lower enrichment and a correspondingly lower required burnup representative of the majority of assemblies.

3. The reactivity effect due to burnup record uncertainty is determined at the 4.60 wt%/50.58 GWD/MTU level which is not representative of the majority of assemblies to be stored. Most are a much lower enrichment requiring a lower assembly burnup which would result in a smaller reactivity effect.

4. The axial burnup adjustment factors utilized are based on limiting axial profiles for historical Palisades fuel. Not all assemblies are burned to these limiting shapes. Furthermore, current Palisades fuel designs incorporate a reduced enrichment axial blanket on the top and bottom of each fuel pin. The reactivity effects of the axial burnup profile in these assemblies are expected to be much smaller than those considered in this analysis.

## 9.0 COMPUTER CODE BENCHMARKING

MONK is designed to systematically over predicted $k_{eff}$ for enriched $UO_2$ systems when using the UKNDL based cross-section library. An extensive validation database is available to quantify the amount of over prediction. The subset of benchmark experiments consisting of $UO_2$ pins span a range of enrichments from 2.35 wt% to 7.0 wt% $^{235}U$. The over prediction in the MONK calculations ranges from a high of 0.01 $\Delta k$ to 0.0003 $\Delta k$. A categorization facility is available in MONK to assist the engineer in determining the type of system he is assessing. The categorization scheme evaluates seven properties; 1) type of fissile material, 2) non-fuel absorption, 3) leakage, 4) resonance absorption, 5) fast fission, 6) spectrum and, 7) geometry. The main use of the categorization is to enable the user to quickly check that a calculation is adequately covered by validation cases and, more importantly, to immediately alert him/her when a case is not covered by the validation. Table 5 provides an example of the validation database experiments for category 121 systems. [52] Many of the calculations discussed in this report are classified in category 121 by the MONK categorization algorithm. Table 5 gives the calculated $k_{eff}$, experimental $k_{eff}$, and the corresponding standard errors. The MONK User's Guide provides the specific properties of each experiment. The typical MONK over prediction in these type of criticality evaluations is evident from the results in Table 5. Any calculated bias would be negative and the MONK calculated $k_{eff}$ is considered conservative. Therefore the criticality evaluations in this report conservatively make no adjustments to the MONK calculated $k_{eff}$ due to a method bias. The 95/95 $k_{eff}$ is determined using the unbiased value.

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52: An overview of the MONK validation database is provided in ANSWERS/MONK(94)3, MONK User's Guide, Section 7.

Table 5 provides only a small subset of the experimental benchmarks which comprise the validation database for MONK 7A. All benchmark calculations are performed by AEA technology and not reproduced by Consumers Energy. The MONK code installation on the Palisades DEC Alpha computer network is verified through the performance of vendor supplied test cases [53]. Further, AEA was contracted to perform checks of the MONK case outputs which are the basis for the evaluations presented in this report. AEA's review verified that the code was run correctly and that results are interpreted conservatively [54]. In addition, Consumers Energy has compared results from the MONK models of the new fuel storage, Region I and Region II racks to KENO-Va calculations of record [55]. The comparisons support the validity of the respective models and show the conservative over prediction inherent in the MONK UKNDL calculation. Table 6 provides the MONK and KENO calculated $k_{eff}$'s.

MONK 6B has been reviewed by the Nuclear Regulatory Commission [56] [57] and found acceptable for use in criticality studies. The evolution of MONK 7A conserves all of the major features of MONK 6B. The nuclear data library, which is the basic determinant of the accuracy of the code, is retained. New features in MONK 7A serve to increase the code's flexibility. The functional differences between MONK 7A and MONK 6B are [58]:

- A new geometry modeling package that includes the MONK 6B capability as a sub-set. This extends still further the power and versatility of MONK for modeling complex situations.
- A new thermalization treatment for hydrogen when bound in water and poly-carbons. This provides additional physical realism in the MONK collision modeling.
- New starting source options to simplify the specification of an accurate initial fission distribution.
- Revised output format to enable calculations to be more readily interpreted.
- Improved geometry checking and visualization tools.
- Comprehensively updated user documentation.

Extensive validation of MONK 7A has taken place. A comparison of MONK 6B and MONK 7A for four key sets of critical experiments relevant to low-enriched uranium fuel storage and transportation situations is presented in Table 7 [59]. The first three experiments were included in the MONK 6B topical report. The final set has been added as part of a continuing MONK validation program. The validation results given in

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53: EA-OSF-94-05, "*Installation of MONK 7a ...*"

54: AEAT1 and AEAT4

55: EA-SFP-97-02, EA-SFP-97-03, EA-SFP-98-04.

56: AEA RS 5520, "*Topical Report on the Use of MONK 6B ....*"

57: "*Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to the AEA O'Donnell Topical Report AEA RS 5520 ...*"

58: AEA RS 5520 Addendum, "*The Application of MONK 7A ...*"

59: AEA RS 5520 Addendum, "*The Application of MONK 7A ...*"

Table 7 demonstrate that the accuracy of MONK 7A for the analysis of low-enriched uranium systems is statistically consistent with the results obtained for MONK 6B.

The USNRC has approved the use of CASMO-3 by Consumers Energy for the calculation of fuel cross sections utilized in the core monitoring software, PIDAL [60]. Consumers energy has extensive experience with use of CASMO-3/SIMULATE-3 for rod worth calculations, estimated critical boron calculations and core design. These calculations require the modeling of complex arrays of fuel assemblies with differing burnup and initial enrichments. While CASMO is utilized only for the relative change in $k_{eff}$, $\Delta k$, either from the variation in certain manufacturing dimensions or from assembly burnup, the Region I and Region II calculated $k_\infty$ is compared to the MONK and KENO calculated values in Table 6 in order to validate the model. Further, AEA technology was contracted to perform alternate calculations with the WIMS lattice code [61]. A specific uncertainty for the CASMO-3 burnup calculation utilized in the reactivity equivalencing methodology is not rigorously developed by Consumers Energy. Rather, 5% of the largest reactivity effect credited as a result of burnup is utilized. This value, 0.02137, is then conservatively rounded to 0.025.

60: Docket No. 50-255, "*Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Revision of PIDAL...*"

61: AEAT2 and AEAT3

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## 11.0 TABLES

**Table 1** Fuel Parameters.

| Parameter | Nominal Modeled Values |
|---|---|
| Maximum Enrichment | 4.95 |
| Fuel Rod Array | 15x15 |
| Number of Rods/Assembly | 216 |
| Number of Non-fueled Positions/Assembly | 9; 8 guide bars, 1 instrument tube |
| Assembly Nominal Envelope Dimensions | 8.25" x 8.25" |
| Fuel Rod Pitch | 0.5500" |
| Cladding OD | 0.4170" |
| Cladding ID | 0.3670" |
| Cladding Thickness | 0.0250" |
| Fuel Rod Pellet-to Clad Diametrical Gap | 0.0070" |
| Pellet OD | 0.3600" |
| Density | 96.0 % TD |
| Dish Volume | 0% |

**Table 2** Region I Statistical Uncertainty Calculation [62].

| Parameter | | Δk | Δk |
|---|---|---|---|
| Density | | 0.00151 | |
| Pellet Diameter | | 0.00024 | |
| Enrichment | | 0.00156 | |
| Clad ID | | 0.00006 | |
| Clad OD | | 0.00302 | |
| Can Inner Wall | | 0.00033 | |
| Can Outer Wall (+) | | 0.00213 | |
| Can Outer Wall (-) | | 0.00073 | |
| Cell Pitch | | 0.00506 | |
| $B_4C$ Plate Thickness | | 0.00403 | |
| $B_4C$ Plate Width | | 0.00125 | |
| B-10 density | | 0.00050 | |
| | SRSS [illegible] | | 0.007920 |
| | 2 * Monk Standard deviation | | 0.001200 |
| | SRSS [illegible] | | 0.008104 |

**Table 3** Region II Statistical Uncertainty Calculation [63].

| Parameter | | Δk | Δk |
|---|---|---|---|
| Density | | 0.00215 | |
| Pellet Diameter | | 0.00033 | |
| Enrichment | | 0.01588 | |
| Clad ID | | 0.00039 | |
| Clad OD | | 0.00054 | |
| Cell ID | | 0.00268 | |
| Cell Pitch | | 0.00627 | |
| Wall Thickness | | 0.00773 | |
| | SRSS [illegible] | | 0.019068 |
| | 2 * Monk Standard deviation | | 0.000600 |
| | SRSS [illegible] | | 0.01910 |

62: Data taken from EA-SFP-97-02, Section 4.5 , "*Uncertainties*".

63: Data taken from EA-SFP-98-04, Section 4.5, "*Uncertainties*".

Table 4 Region II Burnup Requirements

| Enrichment (wt%) | Burnup (GWD/ MTU) No Decay | Burnup (GWD/ MTU) 1 Year Decay | Burnup (GWD/ MTU) 3 Year Decay | Burnup (GWD/ MTU) 5 Year Decay | Burnup (GWD/ MTU) 8 Year Decay |
|---|---|---|---|---|---|
| | | | | | |
| 1.14 | 0 | 0 | 0 | 0 | 0 |
| 1.14 | 3.477 | 3.477 | 3.477 | 3.477 | 3.477 |
| 1.20 | 3.477 | 3.477 | 3.477 | 3.477 | 3.477 |
| 1.40 | 7.951 | 7.844 | 7.464 | 7.178 | 6.857 |
| 1.60 | 11.615 | 11.354 | 10.768 | 10.319 | 9.847 |
| 1.80 | 14.936 | 14.535 | 13.767 | 13.187 | 12.570 |
| 2.00 | 18.021 | 17.502 | 16.561 | 15.875 | 15.117 |
| 2.20 | 21.002 | 20.417 | 19.313 | 18.499 | 17.611 |
| 2.40 | 23.900 | 23.201 | 21.953 | 21.034 | 20.050 |
| 2.60 | 26.680 | 25.905 | 24.497 | 23.487 | 22.378 |
| 2.80 | 29.388 | 28.528 | 27.006 | 25.879 | 24.678 |
| 3.00 | 32.044 | 31.114 | 29.457 | 28.243 | 26.942 |
| 3.20 | 34.468 | 33.457 | 31.698 | 30.397 | 29.008 |
| 3.40 | 36.848 | 35.783 | 33.920 | 32.544 | 31.079 |
| 3.60 | 39.152 | 38.026 | 36.059 | 34.615 | 33.077 |
| 3.80 | 41.419 | 40.226 | 38.163 | 36.650 | 35.049 |
| 4.00 | 43.661 | 42.422 | 40.257 | 38.673 | 37.007 |
| 4.20 | 45.987 | 44.684 | 42.415 | 40.778 | 39.028 |
| 4.40 | 48.322 | 46.950 | 44.588 | 42.877 | 41.041 |
| 4.60 | 50.580 | 49.158 | 46.690 | 44.911 | 43.003 |
| | | | | | |

**Table 5** Example MONK 7AValidation Calculations.

| Case Number | Experiment | Calculated $k_{eff}$ | Standard Error | Experimental $k_{eff}$ | Standard Error |
|---|---|---|---|---|---|
| 2.01 | 4.75 wt% $^{235}U$ Enriched pins 1.26 cm pitch | 1.0103 | 0.0011 | 1.0000 | 0.0040 |
| 2.04 | 4.75 wt% $^{235}U$ Enriched pins 1.35 cm triangular pitch | 1.0097 | 0.0011 | 1.0000 | 0.0040 |
| 2.07 | 4.75 wt% $^{235}U$ Enriched pins 1.26 cm pitch | 1.0109 | 0.0011 | 1.0000 | 0.0040 |
| 7.01 | 2.46 wt% $^{235}U$ Enriched Pins | 1.0103 | 0.0011 | 1.0000 | 0.0020 |
| 7.02 | 2.46 wt% $^{235}U$ Enriched Pins | 1.0071 | 0.0011 | 1.0000 | 0.0020 |
| 7.03 | 2.46 wt% $^{235}U$ Enriched Pins | 1.0068 | 0.0011 | 1.0000 | 0.0020 |
| 7.04 | 2.46 wt% $^{235}U$ Enriched Pins | 1.0083 | 0.0011 | 1.0000 | 0.0020 |
| 7.05 | 2.46 wt% $^{235}U$ Enriched Pins | 1.0040 | 0.0011 | 1.0000 | 0.0020 |
| 7.06 | 2.46 wt% $^{235}U$ Enriched Pins | 1.0026 | 0.0011 | 1.0000 | 0.0020 |
| 27.01 | 7% Fuel Lattice | 1.0074 | 0.0010 | 1.00000 | 0.0020 |

**Table 6** Comparison of KENO, MONK and CASMO Calculations.

| | CASMO-3 | MONK 7A | KENO Va |
|---|---|---|---|
| New Fuel Storage (nominal) [64] | N/A | 0.9344 | 0.92154 |
| New Fuel Storage (adverse) [65] | N/A | 0.9428 | 0.93417 |
| Region I [66] | 0.9043 | 0.9051 | 0.9006 [67] |
| Region II [68] | 0.9344 | 0.9458 | 0.9390 |

64: EA-SFP-97-03 Table 4.6 and Table 4.7.

65: EA-SFP-97-03 Table 4.6 and Table 4.7.

66: EA-SFP-97-01 Table 4.6 and EA-SFP-97-02 Table 4.3.

67: KENO-IV 95/95 value. The Calculation Uncertainty component was not provided in the vendor report.

68: EA-SFP-98-03 Table 4.6 and EA-SFP-98-04 Table 4.7.

**Table 7** Comparison of MONK 7A to MONK 6B [69].

| **Description of Experiment** | **# of Cases** | **Mean $k_{eff}$ (Measured)** | **Mean $k_{eff}$ (MONK 6B)** | **Mean $k_{eff}$ (MONK 7A)** |
|---|---|---|---|---|
| Clusters of 2.35 wt% enriched $UO_2$ fuel - water reflected - various absorbing plates | 9 | 1.0000±0.0023 | 1.0046±0.0004 | 1.0036±0.0003 |
| Clusters of 4.75 wt% enriched $UO_2$ fuel - water reflected - various moderation levels | 8 | 1.0000±0.0021 | 1.0093±0.0007 | 1.0089±0.0006 |
| Clusters of 4.31wt% enriched $UO_2$ fuel - water reflected - various absorbing plates | 9 | 1.0000±0.0017 | 1.0034±0.0006 | 1.0026±0.0005 |
| Arrays of 2.46 wt% enriched $UO_2$ fuel - water reflected - various absorbing pins and plates | 8 | 1.0000±0.0020 | 1.0058±0.0008 | 1.0050±0.0010 |

69: All results taken from AEA RS 5520 Addendum, "*The Application of MONK 7A ...*"

Intentionally Blank

## 12.0 FIGURES

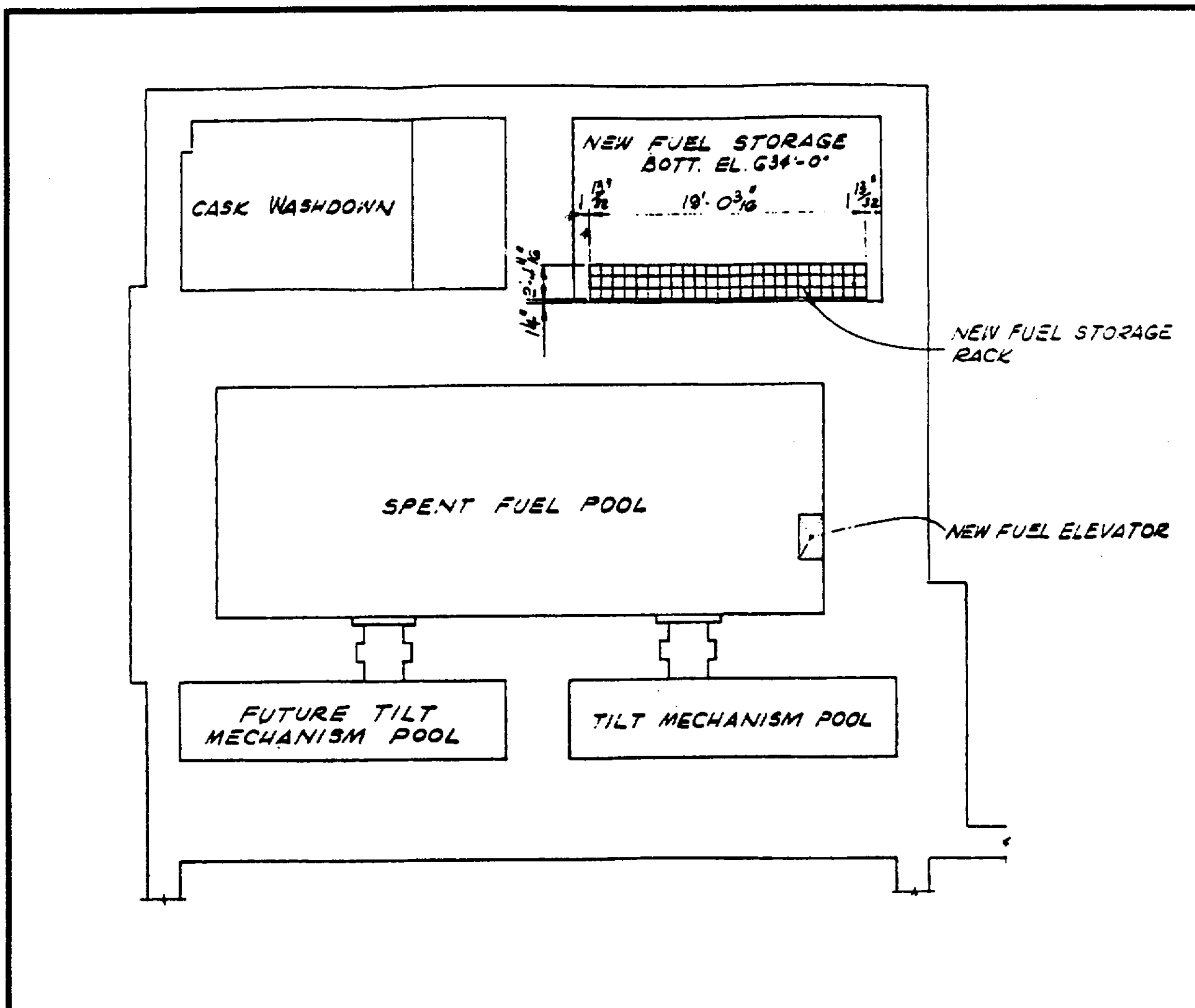


**Figure 1** Layout of Palisades Fuel Pool Area (Bottom is West)

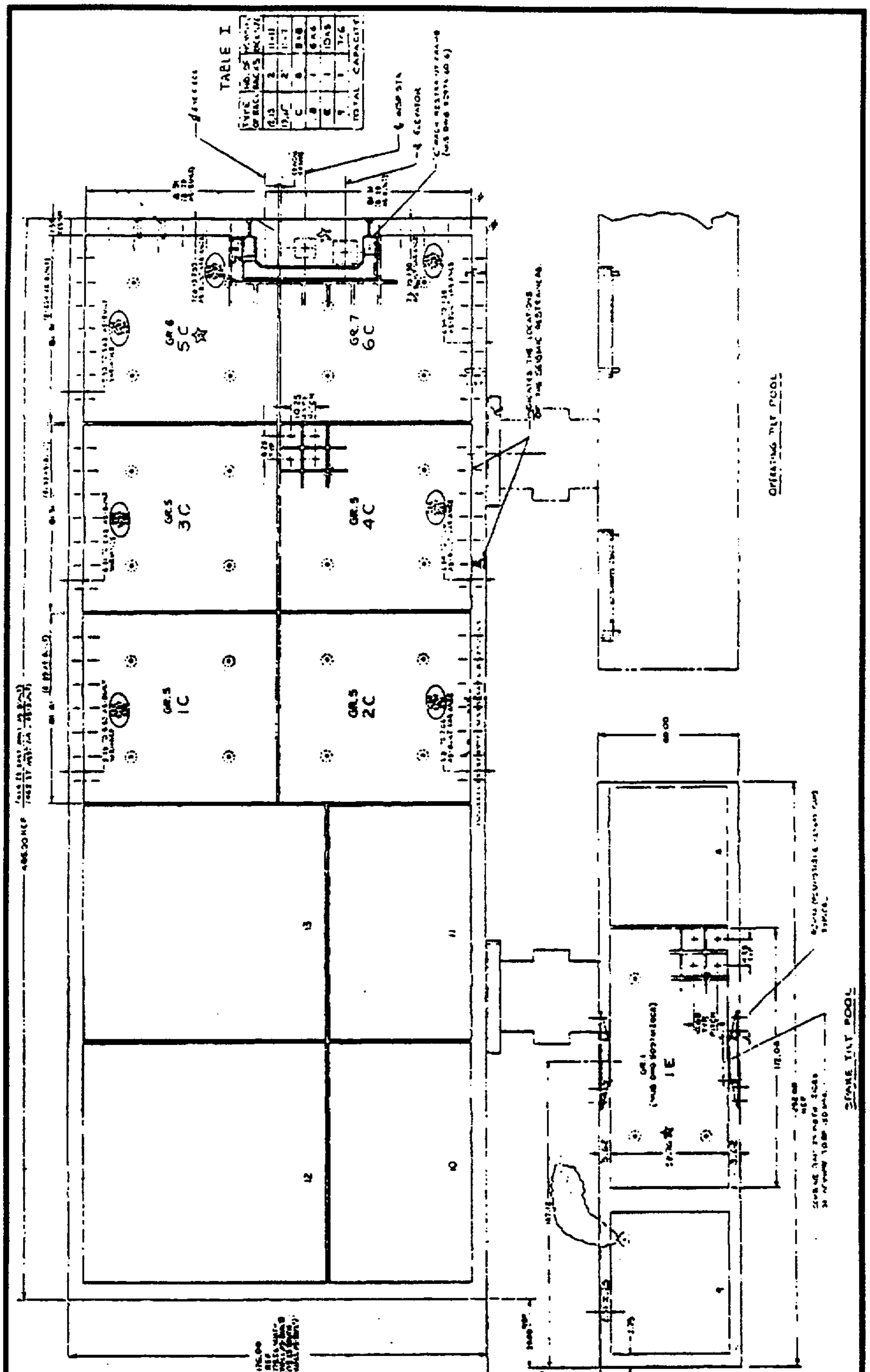

**Figure 2** Palisades Fuel Pool (Bottom is North).

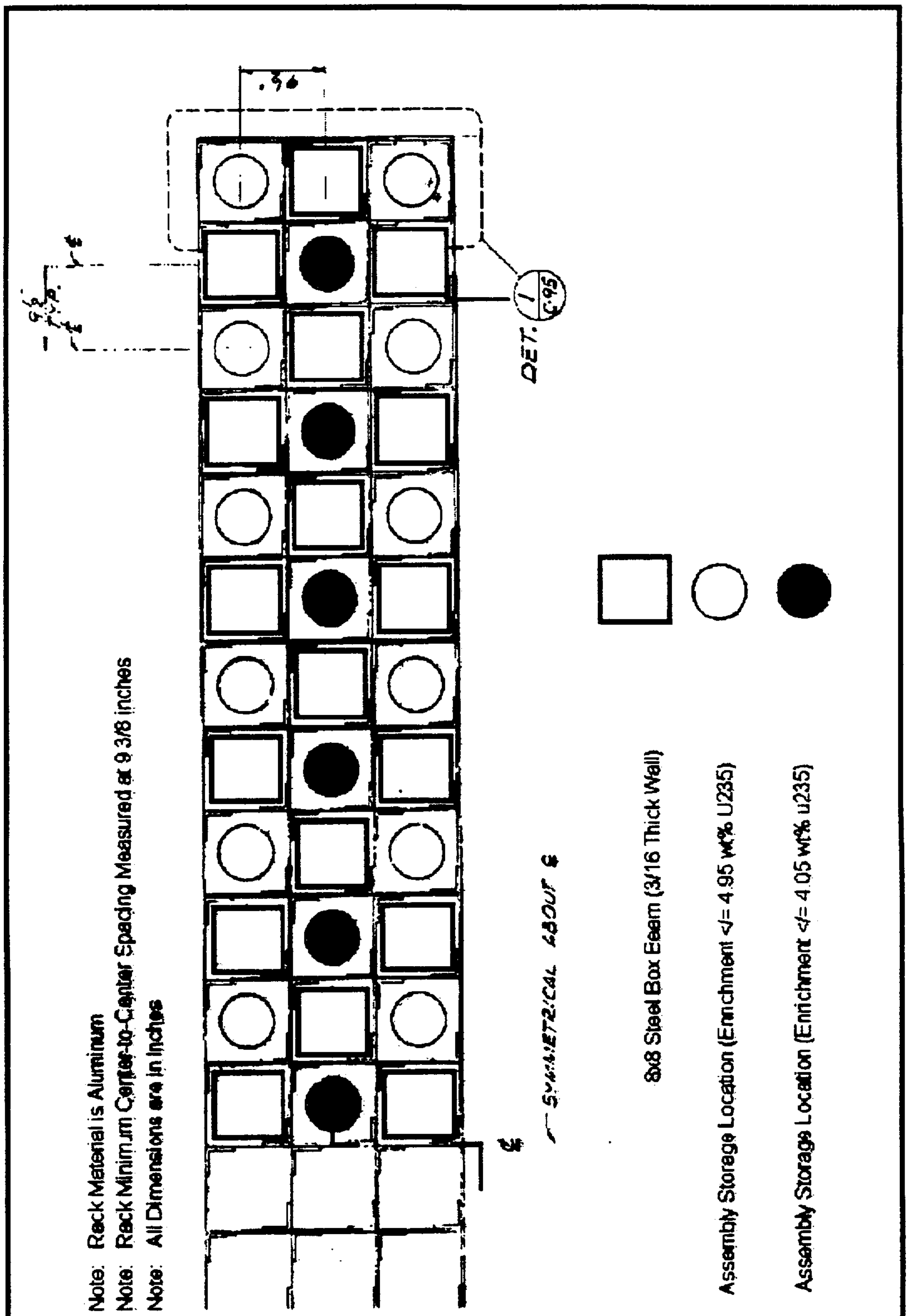


**Figure 3** Palisades New Fuel Storage Rack Loading Paterns.

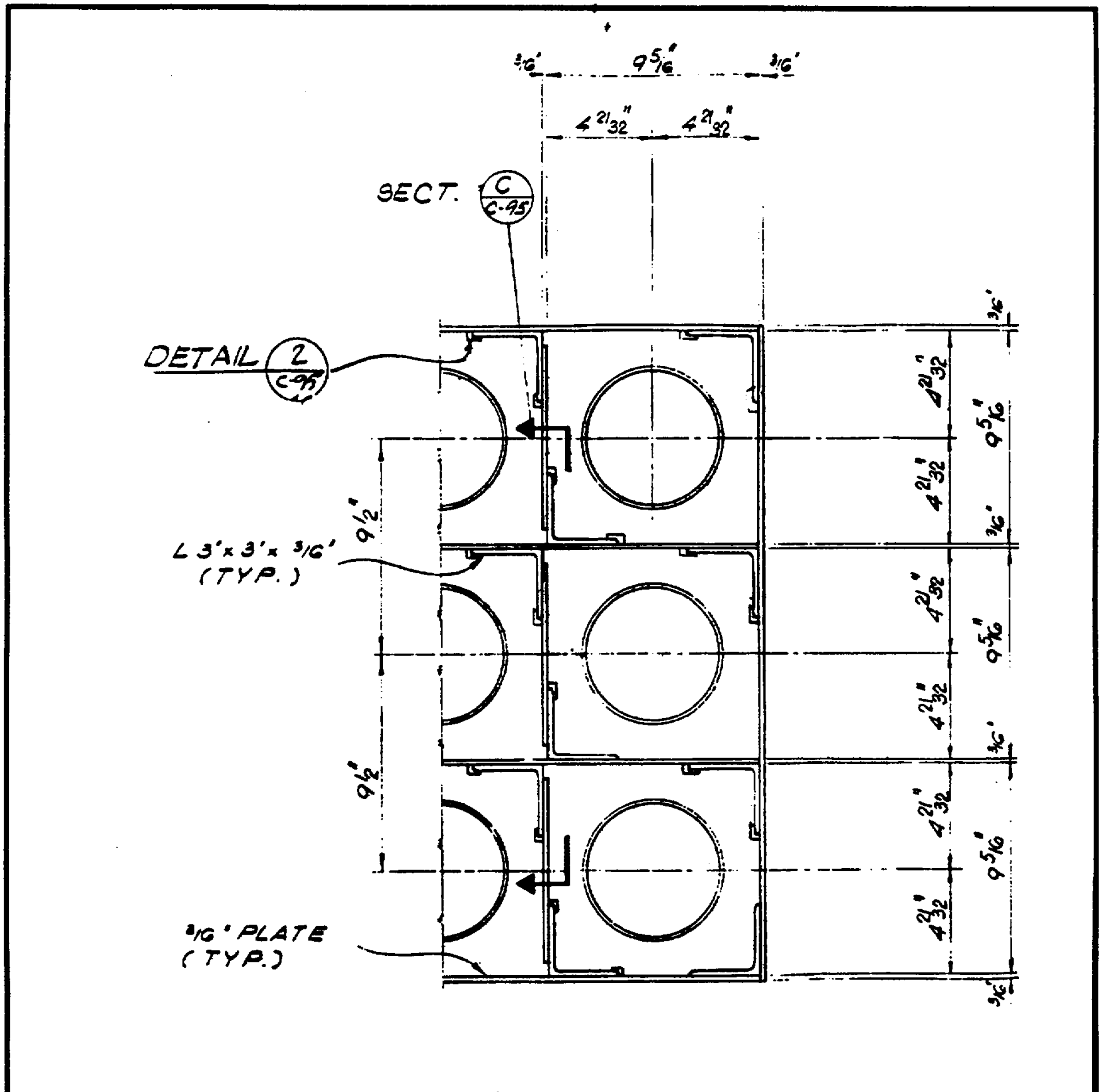


**Figure 4** Palisades New Fuel Storage Rack Cell Details.

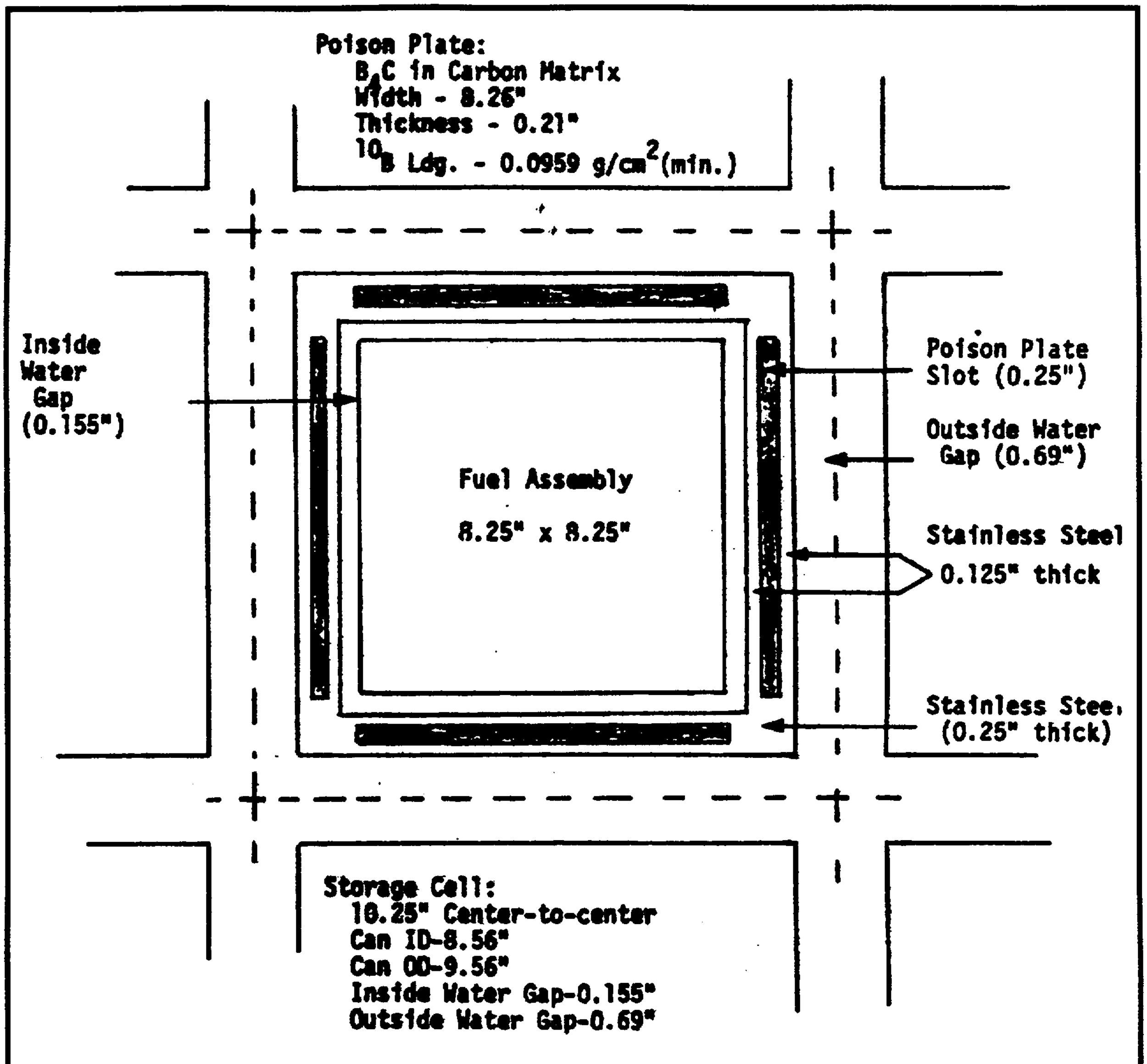


**Figure 5** Palisades Region I (NUS) Main Fuel Pool Storage Rack.

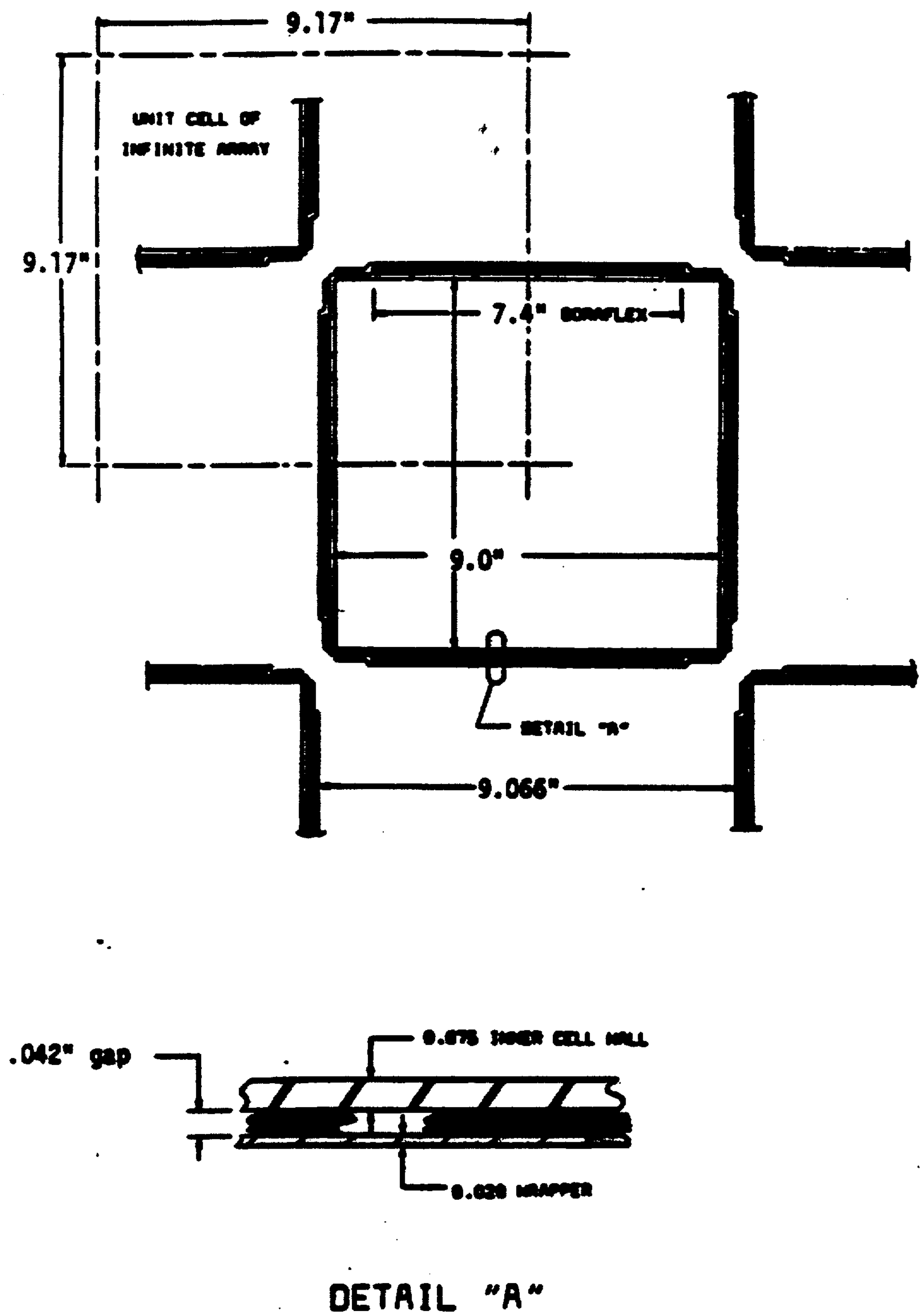


**Figure 6** Palisades Region II (Westinghouse) Fuel Pool Storage Rack.

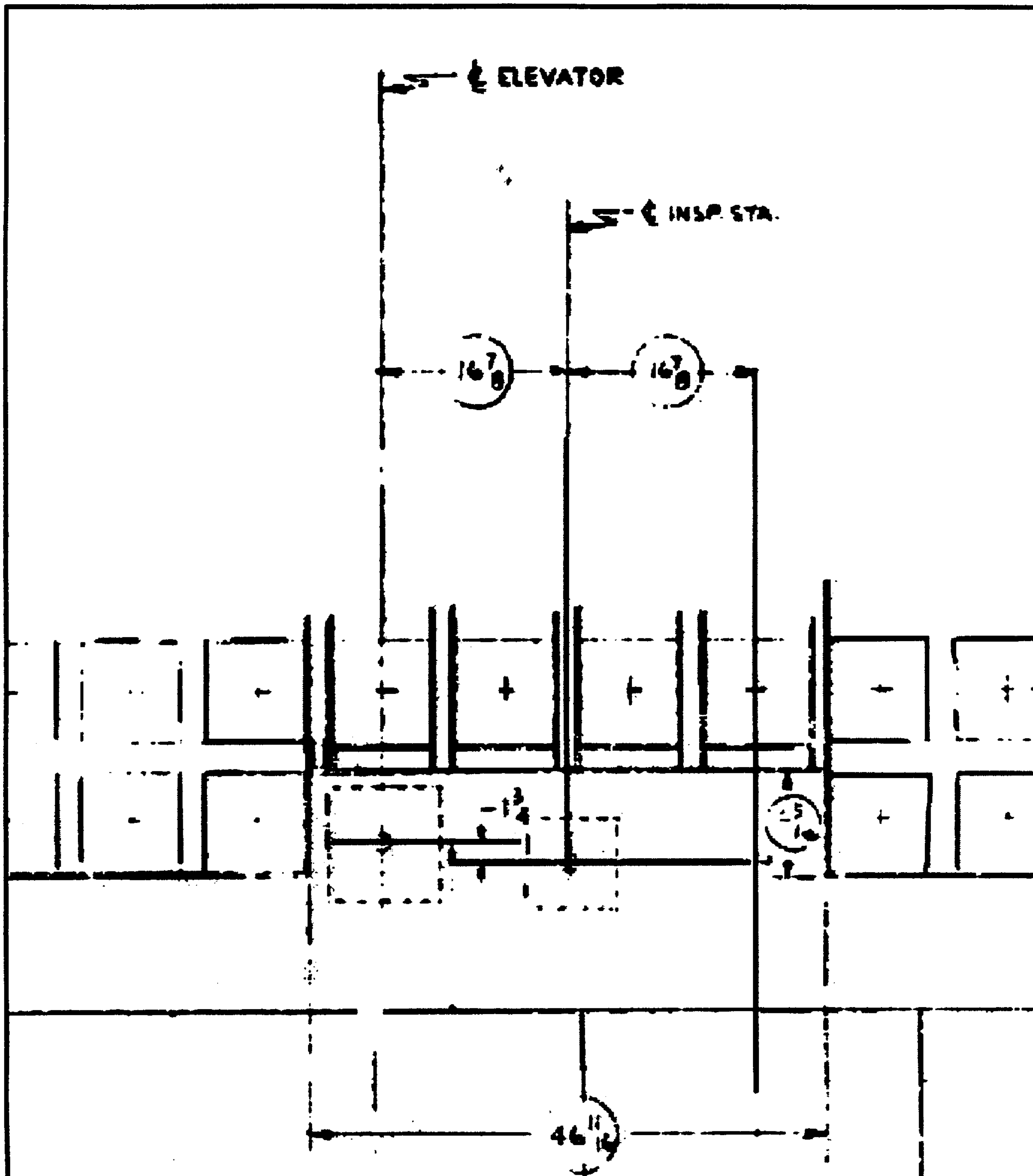


**Figure 7** Palisades Elevator and Inspection Station.

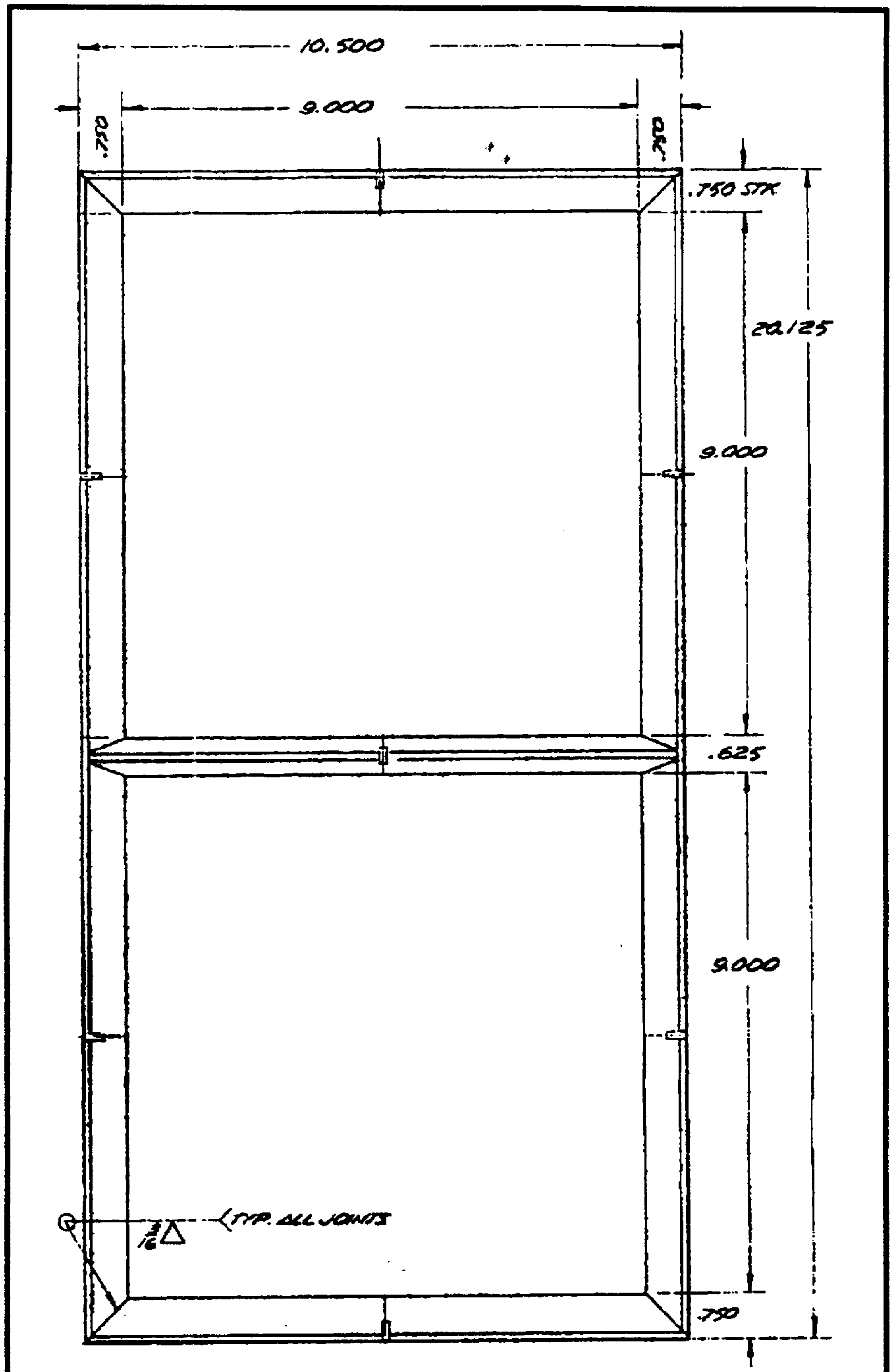


**Figure 8** Palisades Tilt Machine Assembly Cavities.

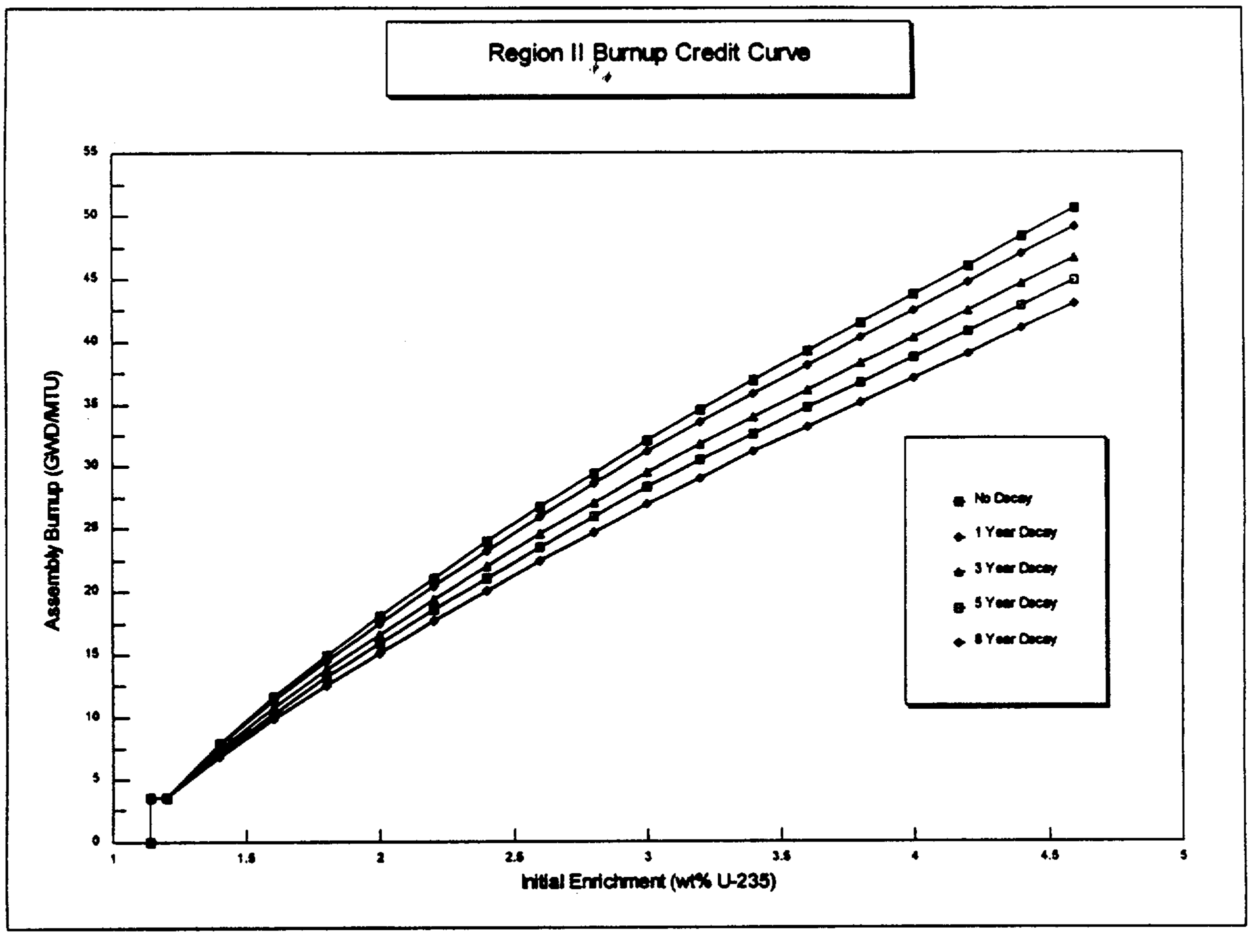


**Figure 9 Region II Burnup Curves.**

**PALISADES NUCLEAR PLANT**
**ENGINEERING ANALYSIS CHECKLIST**

EA -SFP-99-03 REV 0

| SECTION I Items Affected By This EA | Affected Yes | Affected No | Revision Required | Identify* | Closeout |
|---|---|---|---|---|---|
| 1.0 Other EAs | ☑ | ☐ | Supersede | EA-GLP-93-01 | |
| 2.0 Design Documents Electrical E-38 through E-49 | ☐ | ☑ | | | |
| 3.0 Design Documents Mechanical M239-M246, M249, M257-M261, M660, M664-M666 | ☐ | ☑ | | | |
| 4.0 LICENSING DOCUMENTS | | | | | |
| 4.1 Final Safety Analysis Report (FSAR) | ☑ | ☐ | Yes | Section 9.11 | |
| 4.2 Technical Specifications | ☑ | ☐ | | TS 3.7.15, 3.7.16, 3.9.1, 4.3 | |
| 4.3 Operating Requirements Manual | ☑ | ☐ | No → Basis Change | 3.17 Basis | |
| 5.0 PROCEDURES | | | | | |
| 5.1 Administrative Procedures | ☐ | ☑ | | | |
| 5.2 Operating Procedures (SOP, EOP, ONP, etc) | ☑ | ☐ | Yes | ONP 23.3, SOP 27, SOP 28 GOP-11, GOP-2?, GOP-2 | |
| 5.3 Working Procedures | ☐ | ☑ | | | |
| 5.4 Tech Spec Surveillance Test Procedures | ☐ | ☑ | | | |
| 6.0 OTHER DOCUMENTS | | | | | |
| 6.1 Q-List | ☐ | ☑ | | | |
| 6.2 Plant Drawings | ☐ | ☑ | | | |
| 6.3 Equipment Data Base | ☐ | ☑ | | | |
| 6.4 Spare Parts (Stock/MMS) | ☐ | ☑ | | | |
| 6.5 Fire Protection Program Report (FPPR) | ☐ | ☑ | | | |
| 6.6 Design Basis Documents | ☑ | ☐ | | DBD 2.07, DBD 1.10 | |
| 6.7 Operating Checklists | ☐ | ☑ | | | |
| 6.8 SPCC/PIPP Oil and Hazardous Material Spill Prevention Plan | ☐ | ☑ | | | |
| 6.9 EQ Documents | ☐ | ☑ | | | |
| 6.10 MOV/AOV Program Documents (Voltage, thrust, weak link, etc) | ☐ | ☑ | | | |
| 6.11 Work Instructions | ☐ | ☑ | | | |
| 6.12 Other ______ | ☐ | ☑ | | | |

**SECTION II**

Do any of the following documents need to be generated as a result of the conclusions reached in this EA:

| | | Yes | No | Reference |
|---|---|---|---|---|
| 1. | Corrective Action Document? | ☐ | ☑ | |
| 2. | EQ Evaluation Sheet? | ☐ | ☑ | |
| 3. | Safety Evaluation? | ☑ | ☐ | SDR 2000-1380 |
| 4. | Design Basis Document Change Request? | ☑ | ☐ | |
| 5. | FSAR Change Request? | ☑ | ☐ | |
| 6. | Verification Test Procedure (for changes to the Design Basis)? | ☐ | ☑ | |

Completed By *Rob Kaslukovich* Date 10/2/2000

Technical Reviewed By *G E Julles* Date 10/24/00

*Identify Section, No, Drawing, Document, etc.

# TECHNICAL REVIEW CHECKLIST

EA - SFP-99-03 REV 0

This checklist provides guidance for the review of engineering analyses. Answer questions Yes or No, or N/A if they do not apply. Document all comments on a EA Review Sheet. Satisfactory resolution of comments and completion of this checklist is noted by the Technical Review signature at the bottom of this sheet.

(Y, N, N/A)

1. Have the proper input codes, standards and design principles been specified? — Y
2. Have the input codes, standards and design principles been properly applied? — Y
3. Are all inputs and assumptions valid and the basis for their use documented? — Y
4. Is Vendor information used as input addressed correctly in the analysis? — N/A
5. If the analysis argument departs from Vendor Information/ Recommendations, is the departure justification documented? — N/A
6. Are assumptions accurately described and reasonable? — Y
7. Are the design basis changes permitted by this EA bounded by the applicable Safety Review/Evaluation? — Y
8. Are all constants, variables and formulas correct and properly applied? — Y
9. Have all comments been documented on an EA Review Sheet and resolved, or have any minor (insignificant) errors been identified and their insignificance justified? (Indicate "No Comments," if none were made.) — Y
10. If the analysis involves welding, is the following information accurately represented on the analysis drawing (Output document)? — N/A
    - Type of Weld
    - Size of Weld
    - Material Being Joined
    - Thickness of Material Being Joined
    - Location of Weld(s)
    - Appropriate Weld Symbology
11. Has the objective of the analysis been met? — Y
12. Have administrative requirements such as numbering, format, and indexing been satisfied? — Y

Technical Reviewer [signature] Date 10/24/00