FEB 0 1 2005

L-PI-04-129
10 CFR 50.90

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage"

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) hereby requests an amendment to the Prairie Island Nuclear Generating Plant (PINGP) licensing basis to revise the spent fuel pool (SFP) criticality analysis methodology. Based on application of this methodology, revisions to TS and TS Bases 3.7.17, "Spent Fuel Pool Storage," and TS 4.3, "Fuel Storage" are proposed. NMC has evaluated these proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

Westinghouse Nuclear Safety Advisory Letter, NSAL-00-015, "Axial Burnup Shape Reactivity Bias", informed NMC that the current methodology for SFP criticality analysis may be non-conservative with respect to the axial reactivity bias used to account for three-dimensional burnup effects in the two-dimensional model. In response to NSAL-00-015, NMC performed an operability assessment for the PINGP SFP and determined sufficient margin exists to allow continued safe operation of the plant. In a letter from the NRC to Westinghouse, dated July 27, 2001, the NRC stated that due to large conservatisms in the methodology they do not view these non-conservatisms as a safety concern. Recently, re-analyses proposed in this LAR were performed which conservatively models the PINGP stored spent fuel.

Exhibit A contains the licensee's evaluation of this LAR. Exhibit B provides a markup of TS and TS Bases pages. Exhibit C provides clean revised TS and TS Bases pages. Exhibit D is Westinghouse Electric Company Calculation CN-WFE-03-40, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis."

NMC requests approval of this LAR by January 27, 2006. Upon NRC approval of this LAR, NMC requests 90 days to implement the associated changes.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on FEB 0 1 2005

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

Exhibits:

A. Licensee's Evaluation
B. Proposed Technical Specification and Bases Changes (markup)
C. Proposed Technical Specification and Bases Changes (retyped)
D. Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis, Westinghouse Electric Company LLC, dated November 11, 2004

**Exhibit A**

**LICENSEE'S EVALUATION**

**License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage"**



## 1.0 DESCRIPTION

This LAR is a request to amend Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2.

The Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of the proposed Spent Fuel Pool (SFP) criticality analyses for PINGP using the Westinghouse Soluble Boron Credit Methodology. NMC also requests review and approval of the proposed changes to TS and TS Bases 3.7.17, "Spent Fuel Pool Storage", and TS 4.3, "Fuel Storage" which are supported by the proposed analyses.

## 2.0 PROPOSED CHANGE

This LAR proposes changes to the PINGP licensing basis by application of new SFP criticality analyses using a revised methodology.

A brief description of the associated proposed TS and TS Bases changes is provided below along with a discussion of the justification for each change. The specific wording changes to the TS and Bases are provided in Exhibits B and C.

**TS Limiting Condition For Operation (LCO) 3.7.17, "Spent Fuel Pool Storage":**
LCO 3.7.17 defines the combination of initial enrichment, burnup and decay time for the least restrictive spent fuel storage configuration. This least restrictive configuration is referred to as the "All-Cell" configuration in the Westinghouse Electric Company, LLC (Westinghouse) analysis entitled, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality

Analysis," Reference 1. The new SFP "All-Cell" criticality analyses assume a single fuel assembly type that bounds all other fuel types. Thus, only a single figure is required in LCO 3.7.17. A new Figure 3.7.17-1 is provided for the "All-Cell" configuration based on the results of the new criticality analyses. Figure 3.7.17-2 and references to it have been deleted in the LCO statement and SR 3.7.17.1.

**TS 4.3, "Fuel Storage":** TS Section 4.3 provides the criteria for PINGP fuel storage including SFP criticality bases and defines more restrictive new and spent fuel storage configurations in the SFP. These more restrictive configurations are referred to as the "3x3 Array" configurations in Reference 1. References to Figure 3.7.17-2 were deleted since this figure was deleted. The new SFP "3x3 Array" criticality analyses assume two fuel assembly types: 1) fuel rods containing gadolinium (shimmed); and 2) fuel rods without gadolinium (unshimmed). These two bound all other fuel types. Thus, only two figures are required in TS 4.3.1. Figures 4.3.1-1 and 4.3.1-2 were revised to define the "3x3 Array" configuration consistent with the assumptions of the new analyses proposed in this LAR. Two new Figures 4.3.1-3 and 4.3.1-4 are provided for the "3x3 Array" configurations based on the results of the new criticality analyses. Figures 4.3.1-5 through 4.3.1-12 and references to them have been deleted. The References Section was updated to replace the SFP criticality calculation with the proposed Westinghouse analyses in Reference 1.

**TS Bases 3.7.17, "Spent Fuel Pool Storage":** Bases 3.7.17 have been revised to support proposed LCO 3.7.17 and incorporate the assumptions and results of Reference 1. These Bases changes are provided for information and are not part of the LAR.

In summary these changes are acceptable because they are supported by the proposed SFP criticality analyses in attached Exhibit D, Reference 1.

## 3.0 BACKGROUND

Spent fuel pool criticality analyses are performed to demonstrate that the spent fuel pool $k_{eff}$ is conservatively predicted to be less than 0.95. On behalf of Westinghouse Owners Group utilities, Westinghouse developed a methodology for performing spent fuel pool criticality analyses which takes credit for soluble boron in the spent fuel pool. This methodology was documented in WCAP-14416-NP-A, Revision 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Reference 2 In 1995, Prairie Island (PI) submitted for NRC review and approval new criticality analyses to take credit for soluble boron in the PI spent fuel pool. The NRC in License Amendments 129/121 dated June 12, 1997 approved these analyses and the methodology. Although not explicitly referenced in the Prairie Island Operating Licenses or the Technical Specifications, Appendix A of the Operating Licenses, these analyses utilized the Westinghouse methodology provided in Reference 2. WCAP-14416 (Ref. 2) is referenced in the PI Updated Safety Analysis Report (USAR).

The methodology in Reference 2 utilizes a two-dimensional model of the spent fuel. To account for axial, or three-dimensional effects, a reactivity "bias" was included in the model. Another utility determined that the axial bias included in WCAP-14416 (Ref. 2) may not adequately account for the three-dimensional effects. Westinghouse performed an investigation on various aspects of the spent fuel pool criticality analyses supported by WCAP-14416 (Ref. 2). As the result of this investigation, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 00-015, Reference 3, to the affected plants. This NSAL notified the nuclear industry, including NMC, that the methodology provided in Reference 2 may be non-conservative with respect to the axial reactivity bias used to account for three-dimensional burnup effects in the two-dimensional model. The NRC also became aware of these nonconservatisms. As stated in a letter dated July 27, 2001, from the NRC to Westinghouse, the NRC staff does not view the nonconservatisms in the calculated biases as a safety concern, because of large conservatisms used in other aspects of the methodology. However, in the July 27, 2001 letter, the NRC staff also stated that

> [a]lthough this approach may lead to sufficient margin to account for the identified non-conservatism(s) on a plant specific basis, it departs from the Westinghouse methodology of WCAP-14416. Therefore, WCAP-14416 can no longer be relied upon as an approved methodology by the NRC staff or the licensees. For future licensing actions, licensees will need to submit plant-specific criticality calculations for spent fuel pool configurations that include technically supported margins.

To remove further consideration of WCAP-14416 (Ref. 2) and NSAL 00-015 (Ref. 3) for PINGP, Westinghouse performed new criticality analyses using a revised methodology, the Westinghouse Soluble Boron Credit Methodology described in Reference 1, that provides Prairie Island Nuclear Generating Plant-specific criticality calculations for spent fuel pool configurations that include technically supported margins. The results of the SFP criticality analyses support revision of LCO 3.7.17 and TS 4.3 which simplifies these Technical Specification requirements. NMC requests the NRC approve the PINGP proposed analyses, using the revised methodology, and the associated proposed TS changes. The NRC previously reviewed and approved the Westinghouse Soluble Boron Credit Methodology for other plants including the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, License Amendment Nos. 154, on September 25, 2002.

## 4.0 TECHNICAL ANALYSIS

PINGP is a two unit plant located on the west bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by NSP and operated by the Nuclear Management Company (NMC). Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC)

in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Prairie Island Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967.

PINGP was not licensed to NUREG-0800, "Standard Review Plan (SRP)."

Spent Fuel Pool and Stored Fuel

The spent fuel storage pool is a two compartment pool with these compartments designated as Pool 1 and Pool 2. Each pool contains spent fuel storage racks for vertical placement of new or spent fuel assemblies. Pool 1 may contain up to 462 storage positions, except when the pool is used for cask laydown. In the latter case, only 266 storage positions are available since 4 storage racks must be removed to accommodate the storage cask. Pool 2 has up to 1120 storage positions.

The storage racks consist of storage tubes interconnected with each other through upper and lower grids which ensure the proper location of the storage tubes on 9.5 inch pitch in both directions. Each storage tube consists of three components: an inner type 304 stainless steel tube, a layer of Boraflex neutron absorbing material, and an outer skin of type 304 stainless steel. The neutron absorber material is believed to be degraded and is therefore not credited in the spent fuel pool criticality analyses.

Pools 1 and 2 are designed to accommodate new or spent fuel of various initial enrichments, burnup, decay times and numbers of gadolinium rods. Specific details of the spent fuel storage system and the fuel that are relevant to the criticality analyses are provided in Exhibit D, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis, Westinghouse Electric Company LLC, dated November 11, 2004", Reference 1. This license amendment request does not propose any physical changes to the spent fuel storage systems or other plant systems which may have an impact on storage of fuel in the SFP. Thus SFP storage events initiated external to the SFP, such as a boron dilution event, have not changed since credit for soluble boron was previously approved in License Amendment Nos. 129/121. Events initiated external to the SFP have not increased in probability, nor have different types of accidents been created, thus they are not re-evaluated in this submittal.

Licensing Basis for SFP Criticality Analyses – Acceptance Criteria

The SFP criticality analyses are required to ensure that the spent fuel pool multiplication factor, $k_{eff}$, is less than 0.95 as recommended by American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983, Reference 4, and

NRC guidance in Nuclear Regulatory Commission Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978, Reference 5. In addition, sub-criticality of the pool ($k_{eff}$ < 1.0) must be assured on a 95/95 (probability/confidence level) basis, without the presence of the soluble boron in the pool. NRC guidelines, based upon an accident condition in which all soluble poison is assumed to have been lost, specify that the limiting $k_{eff}$ of 1.00 be evaluated in the absence of soluble boron.

The double contingency principle discussed in ANSI/ANS-8.1-1983 and the April 1978 NRC letter allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related accidents, boron is dissolved in the pool water. The presence of soluble boron in the PINGP SFP is controlled by LCO 3.7.16, "Fuel Storage Pool Boron Concentration." SR 3.7.16.1 requires verification of boron concentration every 7 days which is consistent with the requirements of NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

Current Method for Criticality Analyses

The current method for PINGP SFP criticality analyses is contained in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996 (Ref. 2). As discussed in NSAL 00-015 (Ref. 3), this methodology may be non-conservative with respect to the axial reactivity bias used to account for three-dimensional burnup effects in the two-dimensional model. Consequently, NMC in this LAR proposes new PINGP SFP criticality analyses utilizing a revised methodology.

Proposed Criticality Analyses

NMC proposes to use the analyses provided in Exhibit D, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis, Westinghouse Electric Company LLC, dated November 11, 2004", (Ref. 1) as the new SFP analyses. A brief description of the proposed analyses and the supporting revised methodology, its use and results for PINGP SFP are provided here. For a more complete description, refer to Exhibit D.

The methodology presented in Exhibit D is employed to assure the criticality safety of the SFPs and to define limits placed on fresh and depleted fuel assembly storage configurations. The analysis methodology employs SCALE-PC, a personal computer version of the SCALE-4.3 code system, and the two-dimensional integral transport code DIT (Discrete Integral Transport) with an ENDF/B-VI neutron cross section library.
The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on specific classes of personal computers. SCALE-PC includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. Benchmarking of SCALE-PC for use in spent fuel rack criticality analyses is described in Exhibit D Section 1.3.2.

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The multigroup cross sections utilized in DIT are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI).

Collectively these codes demonstrate that the acceptance criteria defined in Exhibit D are met. SCALE-PC was used in benchmarking and evaluating the fuel assembly storage configurations. The DIT code is used for simulation of in-reactor fuel assembly depletion.

<u>Basis for Proposed Licensing Basis Changes and TS Revisions</u>

As discussed in Exhibit D, Westinghouse has modeled the PINGP spent fuel racks and their contents and performed evaluations utilizing the criticality methodology discussed above. Two fuel storage configurations, designated “All Cell” and “3x3 Array”, were defined for combinations of empty storage cells, new fuel and depleted fuel with various initial enrichments, burnup, decay time and burnable poison (gadolinium) content. Fuel assemblies have been evaluated for maximum enrichments up to 5.0 weight percent (w/o).

The All Cell storage configuration is least restrictive in that empty storage cells or fuel that meets the initial enrichment, burnup and decay time requirements of proposed TS Figure 3.7.17-1 can be stored in any pattern adjacent to an empty storage cell or any other fuel assembly which meets these criteria. Based on evaluation, the Westinghouse 14x14 Standard fuel assembly was selected to be the design basis fuel assembly to represent discharged All Cell fuel assemblies.

The 3x3 Array is more restrictive in that the fuel assembly or empty location arrangement is defined in a square of three cells by three cells with a fresh assembly or an empty cell in the center storage cell as shown in proposed Figure 4.3.1-2. The fuel in the surrounding eight cells must meet the initial enrichment, burnup and decay time requirements of proposed TS Figure 4.3.1-3 or Figure 4.3.1-4. Two figures are given to account for fresh fuel assemblies with gadolinium, “shimmed”, or without gadolinium, “unshimmed”. Based on evaluation, the Westinghouse 14x14 Optimized fuel assembly (OFA) was selected to be the design basis fuel assembly to represent fresh fuel assemblies in the center location of the 3x3 Array and the Westinghouse 14x14 Standard fuel assembly was selected to be the design basis fuel assembly to represent peripheral discharged fuel assemblies in the 3x3 Array. An empty cell may be used in any location.

The SFP criticality acceptance criteria were met when these fuel storage configurations were evaluated applying the proposed SFP criticality methodology.

As part of demonstrating that the $k_{eff}$ requirements are met, evaluations were performed to determine soluble boron credit requirements. A soluble boron concentration of 730

parts per million (ppm) assures that $k_{eff}$ is less than or equal to 0.95 when accounting for burnup and reactivity depletion uncertainties and postulated accidents. For an occurrence of the postulated accident conditions, the double contingency principle discussed in ANSI/ANS-8.1-1983 and the April 1978 NRC letter (Refs. 4 and 5) can be applied. This states that the analyses are not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for the postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 730 ppm required to maintain $k_{eff}$ less than 0.95) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. Current SFP criticality analyses required 750 ppm to meet $k_{eff}$ requirements (this value does not consider the additional boron required to mitigate accident induced reactivity increases). LCO 3.7.16 requires the spent fuel storage pool boron concentration to be greater than or equal to 1800 ppm whenever fuel assemblies are stored in the SFP.

## Conclusions

NMC in this LAR proposes to replace the current SFP criticality methodology with the methodology presented in Exhibit D. The codes, methods and techniques contained in the methodology are used to satisfy the acceptance criteria on $k_{eff}$. The proposed methodology utilizes industry accepted analysis codes which have been benchmarked for SFP criticality analyses crediting soluble boron.

NMC proposes to revise LCO 3.7.17 and associated Bases and TS 4.3.1 incorporating the proposed analyses. The criticality analyses utilized two storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment. These two proposed spent fuel storage configurations are defined in proposed Figures 3.7.17-1 and 4.3.1-1 through 4.3.1-4. These storage configurations correspond to the “All Cell” and “3x3 Array” configurations discussed in Exhibit D. The resulting Prairie Island spent fuel pool criticality analyses allow for the storage of fuel assemblies with enrichments up to a maximum of 5.0 weight percent U-235 while maintaining $k_{eff} \leq 0.95$ including uncertainties and credit for soluble boron.

The proposed methodology and analyses provide a conservative approach for demonstrating that the SFP will meet acceptance criteria. The proposed TS changes in conjunction with other current TS requirements assure that the spent fuel will remain subcritical during normal and postulated accident conditions. Operation of the Prairie Island Nuclear Generating Plant with these licensing basis changes and revised Technical Specifications will continue to protect the health and safety of the public.

# 5.0 REGULATORY SAFETY ANALYSIS

## 5.1 No Significant Hazards Consideration

The Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   This license amendment proposes to revise the plant licensing basis by: 1) replacing the spent fuel pool criticality analyses; and 2) revising the spent fuel storage Technical Specifications 3.7.17, “Spent Fuel Pool Storage” and 4.3, “Fuel Storage” utilizing the proposed analyses. The proposed Technical Specification revisions allow spent fuel to be stored in different configurations.

   The proposed changes relate to prevention of criticality accidents in the spent fuel pool. Since the current spent fuel pool criticality analyses and Technical Specifications ensure that a criticality accident does not occur, criticality accidents have not been previously evaluated. Likewise the proposed spent fuel pool criticality analyses and Technical Specifications ensure that a criticality accident does not occur. Thus the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

   Events that could cause a criticality accident were evaluated and analyses demonstrated that the current Technical Specification required soluble boron is more than adequate to assure that a criticality accident does not occur. Thus the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

   This license amendment proposes to revise the plant licensing basis by: 1) replacing the spent fuel pool criticality analyses; and 2) revising the spent fuel storage Technical Specifications 3.7.17, “Spent Fuel Pool Storage” and 4.3, “Fuel Storage” utilizing the proposed analyses. The proposed Technical Specification revisions allow spent fuel to be stored in different configurations.

The proposed licensing basis changes do not involve a change in system operation, or procedures involved with the fuel storage system. It does revise the allowable storage configurations. The proposed changes provide a conservative basis for evaluating spent fuel pool criticality and storage of fuel assemblies in a safe configuration which meets criticality evaluation acceptance criteria. There are no new failure modes or mechanisms created through use of the proposed analyses or proposed Technical Specifications. Use of these licensing basis changes for storage of fuel assemblies does not involve any modification in the operational limits of plant systems. There are no new accident precursors generated with use of these licensing basis changes.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

**3. Do the proposed changes involve a significant reduction in a margin of safety?**

Response: No

This license amendment proposes to revise the plant licensing basis by: 1) replacing the spent fuel pool criticality analyses; and 2) revising the spent fuel storage Technical Specifications 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage" utilizing the proposed analyses. The proposed Technical Specification revisions allow spent fuel to be stored in different configurations.

The proposed licensing basis change will result in a conservative calculation of the required spent fuel pool soluble boron concentration for the proposed fuel storage configurations. The current Technical Specification required spent fuel pool boron concentration significantly exceeds the proposed criticality analyses required boron concentration. The proposed analyses demonstrate that the criticality analysis acceptance criteria for the proposed fuel storage configurations are met. The proposed analyses utilize industry accepted analysis codes which have been benchmarked for the spent fuel pool criticality analyses proposed for the Prairie Island Nuclear Generating Plant. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, the Nuclear Management Company concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

## 5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.68, Criticality accident requirements

10 CFR 50.68 states that,

*If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.*

The proposed PINGP TS 4.3 reflects 10 CFR 50.68 criticality accident requirements and the required parameters from the new criticality analyses.

General Design Criteria

The construction of the PINGP was significantly complete prior to issuance of 10 CFR 50, Appendix A, General Design Criteria. The PINGP was designed and constructed to comply with the Atomic Energy Commission General Design Criteria as proposed on July 10, 1967 (AEC GDC) as described in the plant Updated Safety Analysis Report (USAR). AEC GDC 66 provides design guidance for fuel storage criticality considerations.

AEC GDC proposed Criterion 66 – Prevention of Fuel Storage Criticality

*Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configuration shall be emphasized over procedural controls.*

The spent fuel storage system is currently designed to prevent criticality through a combination of physical systems and processes. This license amendment request does not propose changes to the physical systems. This license amendment request does propose new spent fuel pool criticality analyses of the physical system and proposes new process controls for safe fuel storage configurations. The proposed analyses utilize industry accepted analysis codes which have been benchmarked for the spent fuel pool criticality analyses. The proposed analyses demonstrate that criticality is prevented by the physical storage system and the proposed fuel storage configurations.

With the changes proposed in this license amendment request, the requirements of this Criterion continue to be met.

NUREG-0800 Standard Review Plan Section 9.1.2, "Spent Fuel Storage"

The Prairie Island Nuclear Generating Plant is not licensed to the criteria listed in NUREG-0800, and nothing in the proposed amendment is intended to commit Prairie Island Nuclear Generating Plant to the criteria in NUREG-0800.

However, Section 9.1.2 of NUREG-0800 was reviewed for guidance for evaluating the acceptability of this license amendment request. Section 9.1.2 of NUREG-0800 was written for new facilities which do not credit soluble boron. The changes proposed in this license amendment request only relate to the spent fuel pool criticality, which credits soluble boron in the storage pool, and application of the analyses to Technical Specification requirements. No physical changes are proposed with this license amendment request. Thus, NMC did not identify guidance for acceptability of this license amendment request in Section 9.1.2 of NUREG-0800.

Section 9.1.2 of NUREG-0800 applies 10 CFR 50, Appendix A, General Design Criteria (GDC) 2, 4, 5, 61, 62 and 63 as the acceptance criteria for spent fuel storage facilities. The Prairie Island Nuclear Generating Plant construction was significantly complete prior to issuance of these criteria and thus is not committed to meet them. As discussed above, the Prairie Island Nuclear Generating Plant meets AEC GDC proposed Criterion 66.

Regulatory Requirements/Criteria Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 7.0 REFERENCES

1. Westinghouse Electric Company, LLC calculation CN-WFE-03-40, “Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis”, dated November 11, 2004”.

2. WCAP-14416-NP-A, Revision 1, ”Westinghouse Spent Fuel Rack Criticality Analysis Methodology.”

3. Nuclear Safety Advisory Letter (NSAL) 00-015, November 2, 2000.

4. American Nuclear Society, “American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants”, ANSI/ANS-57.2-1983, October 7, 1983.

5. Nuclear Regulatory Commission Letter from B. K. Grimes, “OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications”, April 14, 1978.

Exhibit B

**Proposed Technical Specification and Bases Changes (markup)**

Technical Specification Pages

| | |
|---|---|
| 3.7.17-1 | 4.0-8 |
| 3.7.17-2 | 4.0-9 |
| 3.7.17-3 | 4.0-10 |
| 3.7.17-4 | 4.0-11 |
| 4.0-2 | 4.0-12 |
| 4.0-4 | 4.0-13 |
| 4.0-5 | 4.0-14 |
| 4.0-6 | 4.0-15 |
| 4.0-7 | 4.0-16 |

Bases pages
(for information only)

| | |
|---|---|
| B 3.7.17-1 | B 3.7.17-6 |
| B 3.7.17-2 | B 3.7.17-7 |
| B 3.7.17-3 | B 3.7.17-8 |
| B 3.7.17-4 | B 3.7.17-9 |
| B 3.7.17-5 | |

## 3.7 PLANT SYSTEMS

### 3.7.17 Spent Fuel Pool Storage

LCO 3.7.17 The combination of initial enrichment, burnup and decay time of each fuel assembly stored in the spent fuel pool shall be within the Unrestricted Region of Figure 3.7.17-1 ~~or Figure 3.7.17-2, as applicable,~~ or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 -------------NOTE------------- LCO 3.0.3 is not applicable. --------------------------------- <br><br>Initiate action to move the noncomplying fuel assembly to an acceptable location. | <br><br>Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.17.1 | Verify by administrative means the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.17-1 or ~~Figure 3.7.17-2, as applicable, or~~ Specification 4.3.1.1. | Prior to storing or moving the fuel assembly |
| SR 3.7.17.2 | Verify spent fuel pool inventory. | Within 7 days after completion of a spent fuel pool fuel handling campaign |

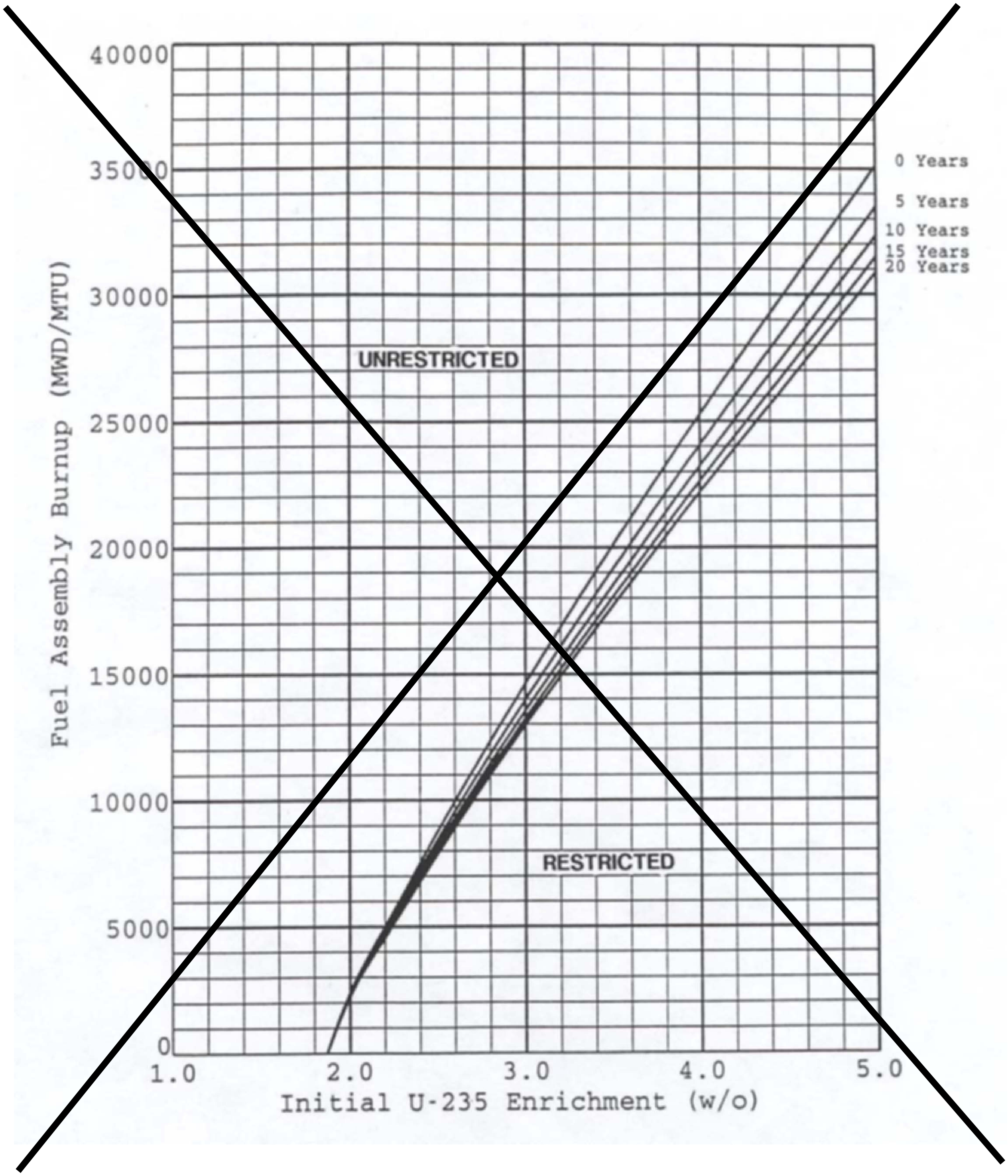
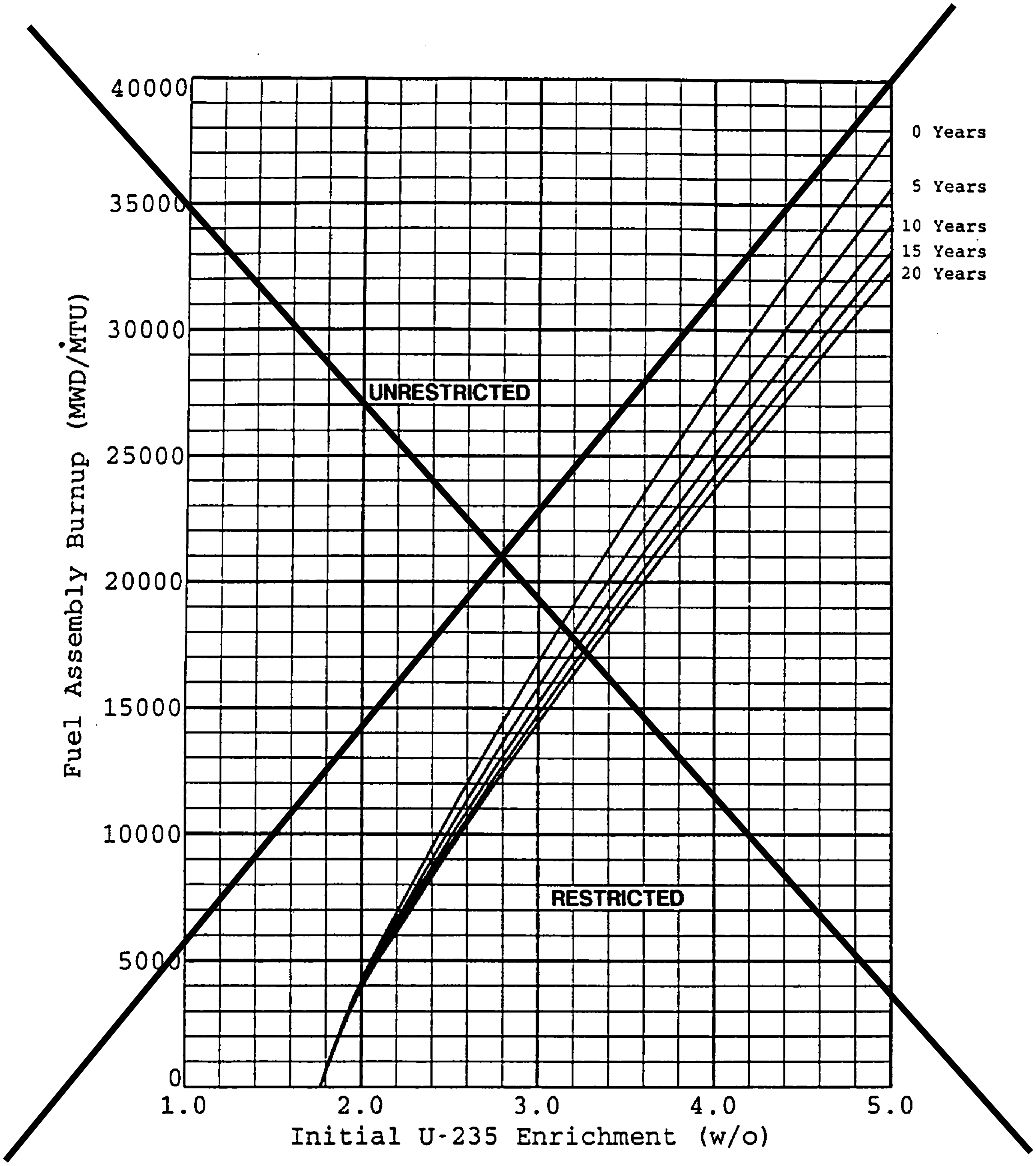



Figure 3.7.17-1
Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements ~~OFA Fuel~~
**Use Figure based on Westinghouse Calculation CN-WFE-03-40 Figure 4-5**




~~Figure 3.7.17-2~~

~~Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements-STD Fuel~~

**Delete this Figure**

## 4.0 DESIGN FEATURES (continued)

### 4.3 Fuel Storage

#### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 1;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to **730**~~750~~ ppm, which includes an allowance for uncertainties as described in Reference 1;

d. A nominal 9.5 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "unrestricted range" of Figure 3.7.17-1~~ or Figure 3.7.17-2, as applicable,~~ may be allowed unrestricted storage in the fuel storage racks; and

f. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "restricted range" of Figure 3.7.17-1~~ or Figure 3.7.17-2, as applicable,~~ will be stored in compliance with Figures 4.3.1-1 through 4.3.1-4~~12~~.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.3 Capacity

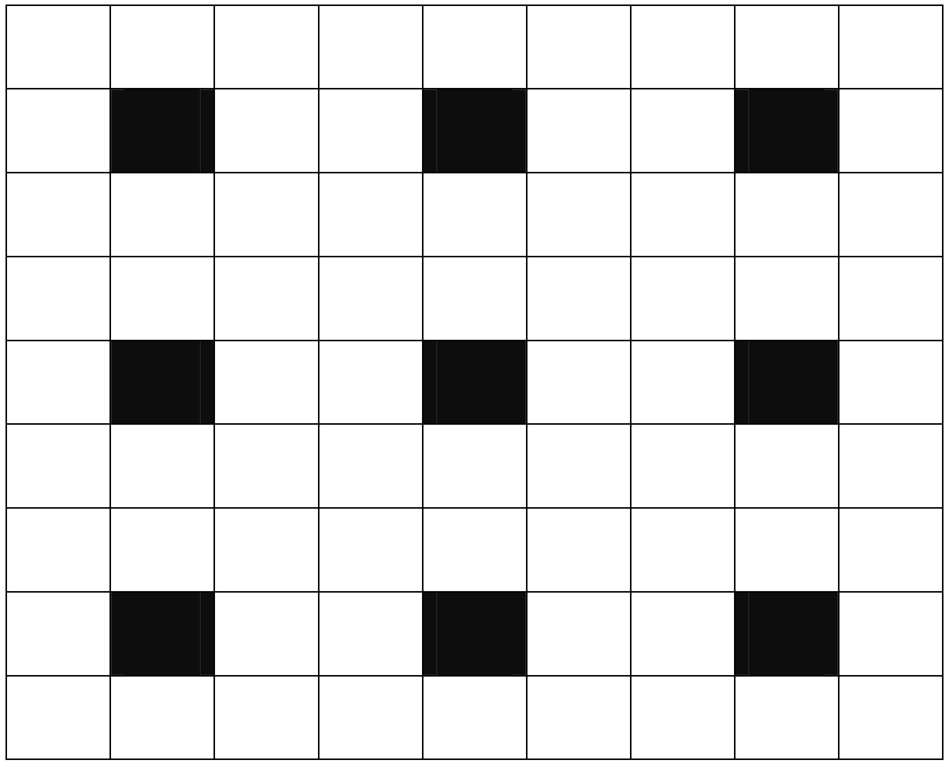
The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies not including those assemblies which can be returned to the reactor. The southeast corner of the small pool serves as the spent fuel cask lay down area. To facilitate plant evolutions, four additional storage racks, with a combined capacity of 196, may be temporarily installed in the cask lay down area to provide a total of 1582 storage locations (Ref. 3).

REFERENCES

1. **"Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis", Calculation Note Number CN-WFE-03-40, Westinghouse Electric Company, November 11, 2004**~~"Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997~~.

2. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

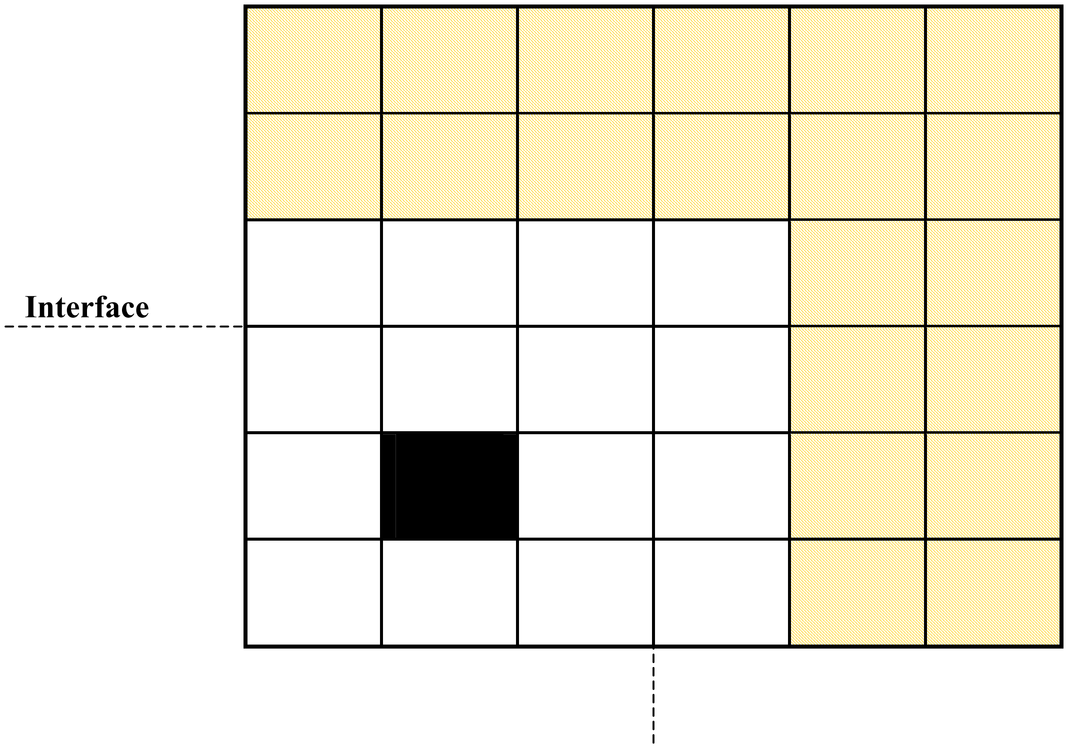
3. USAR, Section 10.2.



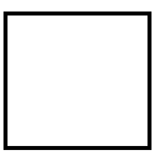
Fresh Fuel: Must be less than or equal to Nominal 4.95 w/o $^{235}U$
No restrictions on burnup

Burned Fuel: Must satisfy minimum burnup requirements of Figures 4.3.1-3 **or** ~~through~~ 4.3.1-**4** ~~12~~ depending on ~~number~~ **presence** of GAD rods in fresh fuel

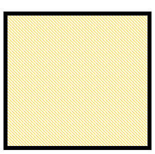
F**igure** ~~IGURE~~ 4.3.1-1
Spent Fuel Pool Burned/Fresh Checkerboard Cell Layout




Fresh Fuel: Must be less than or equal to nominal 4.95 w/o $^{235}U$
No restrictions on burnup

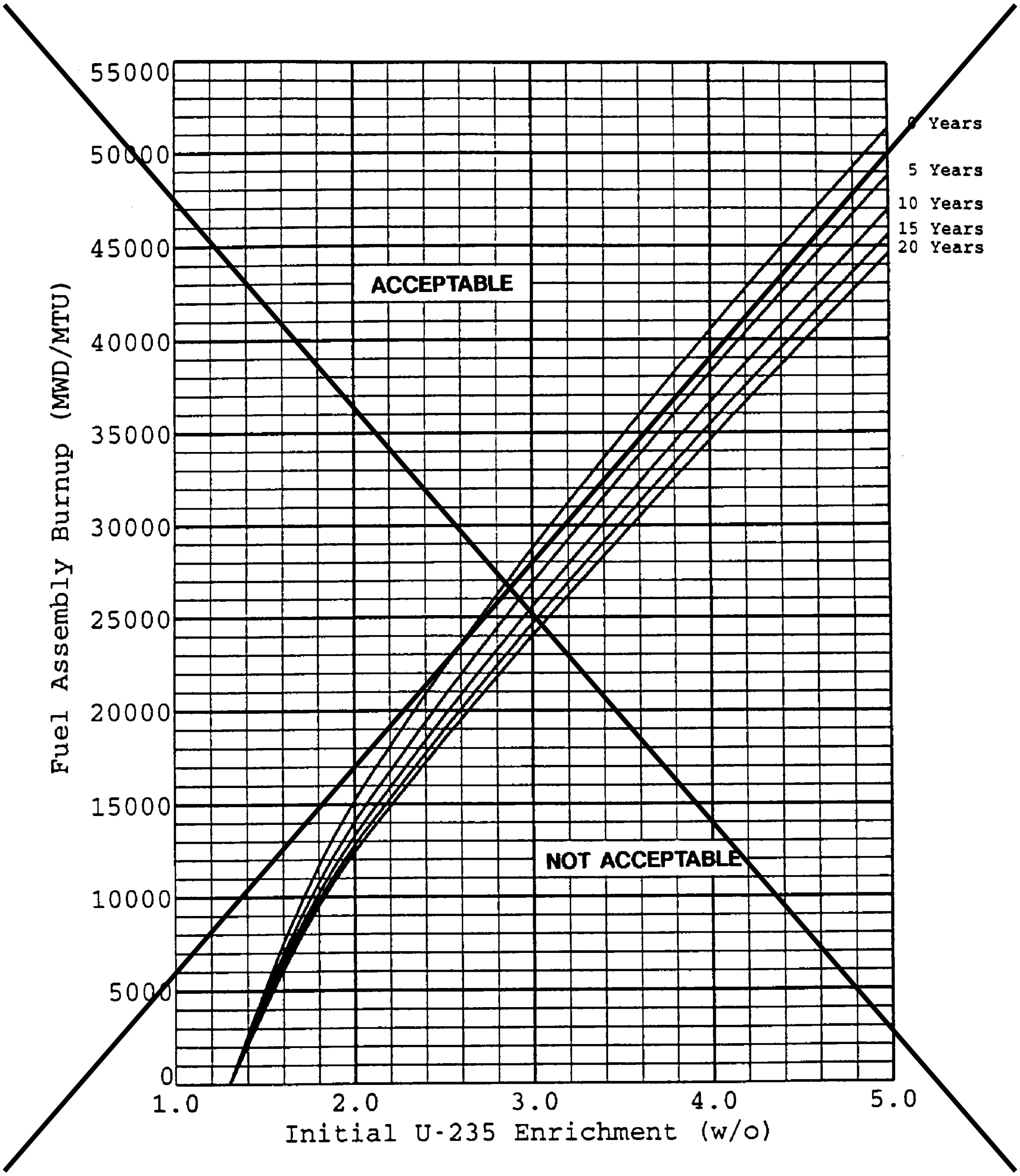


Burned Fuel: 3x3 Checkerboard Region
Must satisfy minimum burnup requirements
of Figures 4.3.1-3 **or** ~~through~~ 4.3.1-**4** ~~12~~



Burned Fuel: All Cell Unrestricted Region
Must satisfy minimum burnup requirements
of Figure~~s~~ 3.7.17-1 ~~or 3.7.17-2~~

F**igure** ~~IGURE~~ 4.3.1-2
Spent Fuel Pool Checkerboard Interface Requirements




**Use Figure based on Westinghouse Calculation CN-WFE-03-40 Figure 4-6**

Figure ~~IGURE~~ 4.3.1-3

Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - ~~OFA Fuel,~~ No GAD

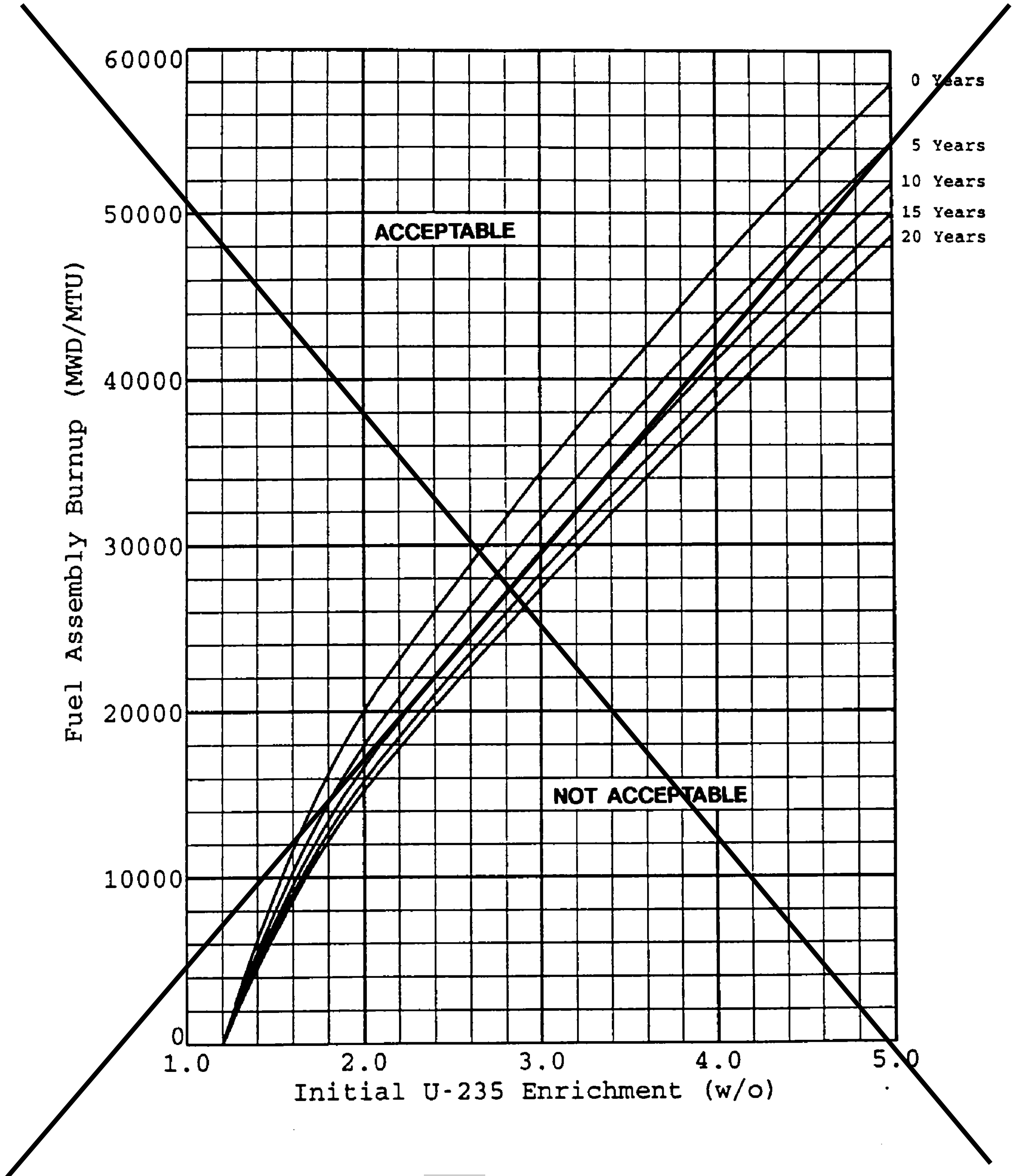
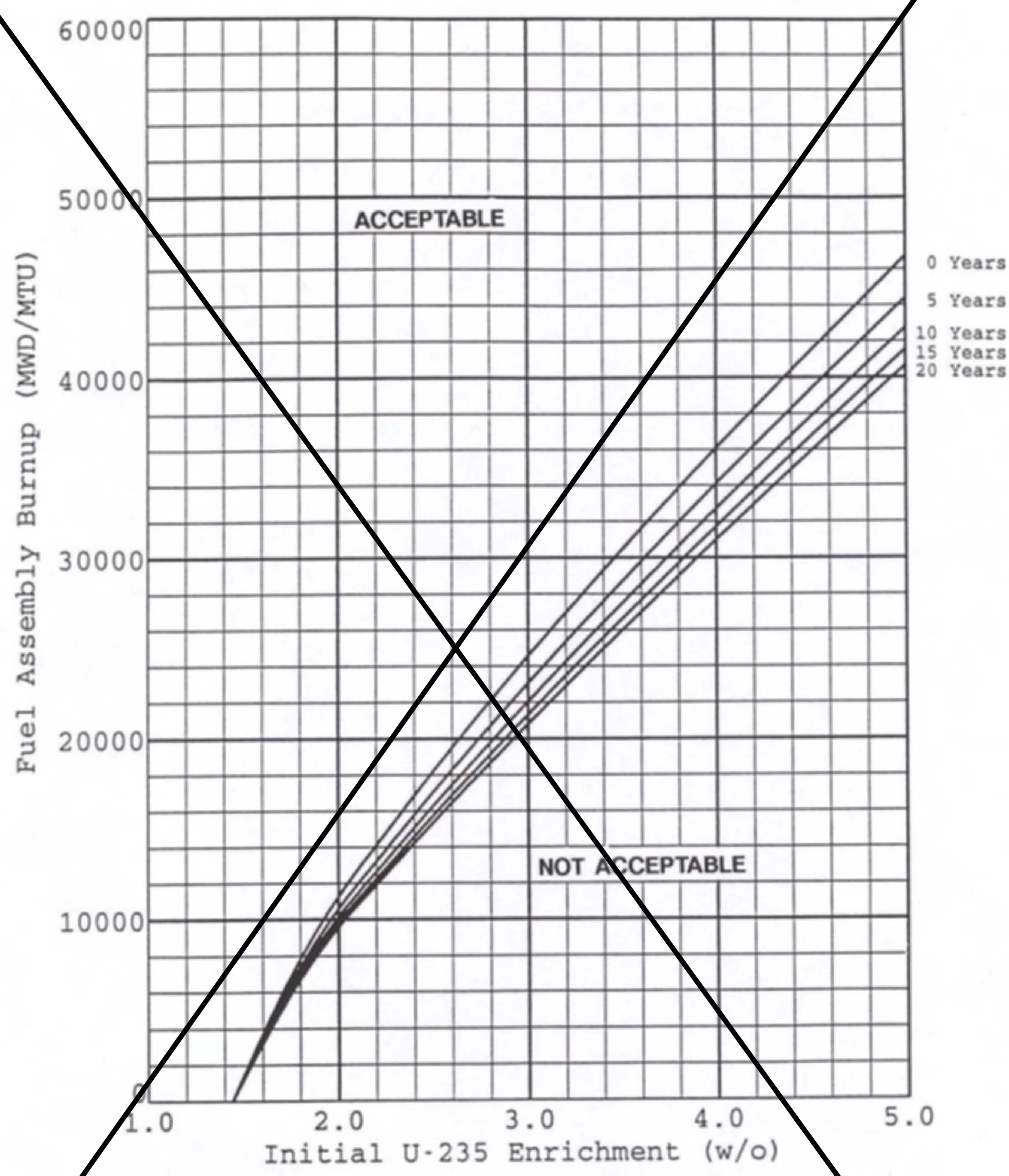
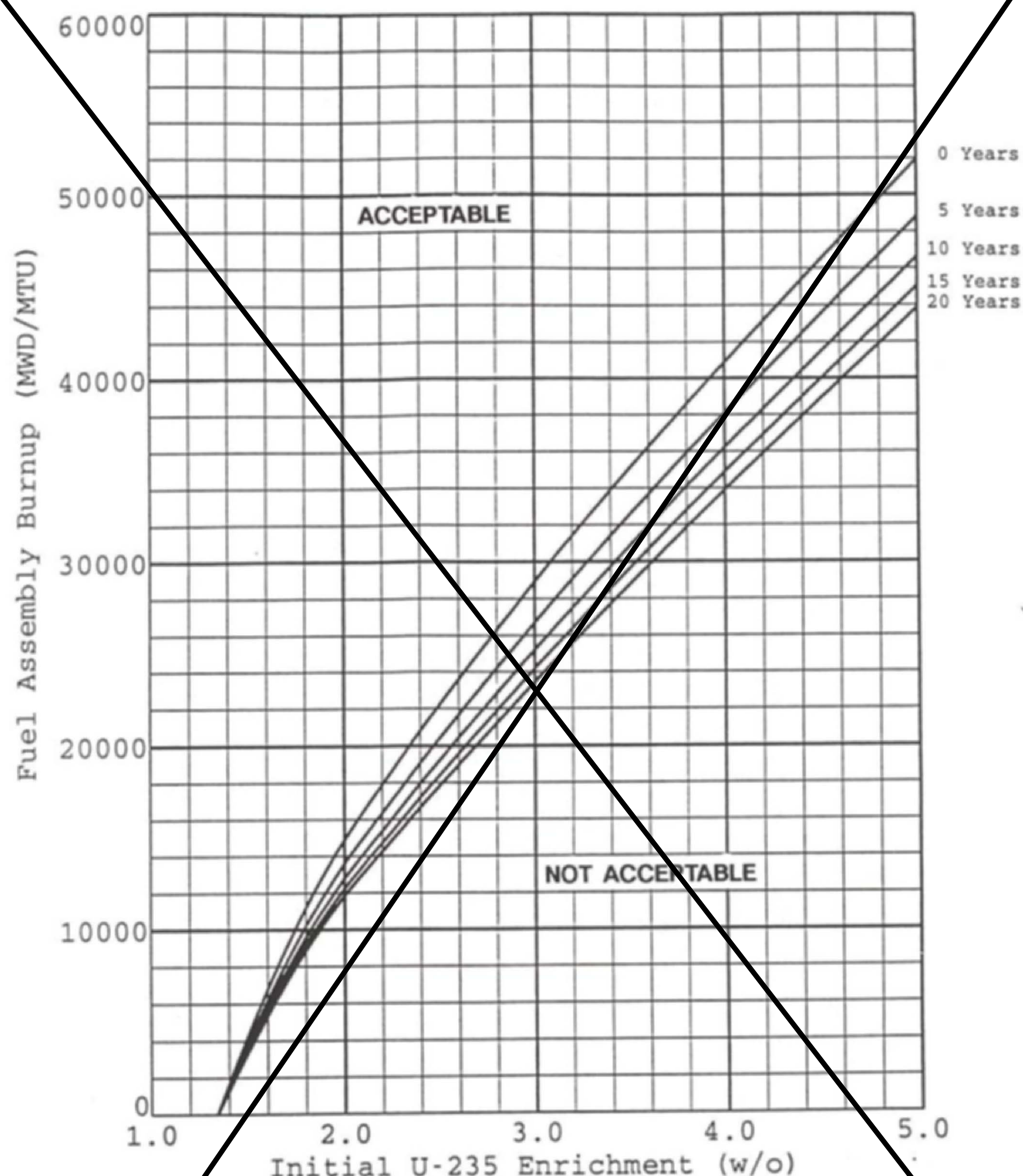



Figure ~~IGURE~~ 4.3.1-4
Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - ~~STD~~ Fuel~~,~~ with ~~No~~ GAD



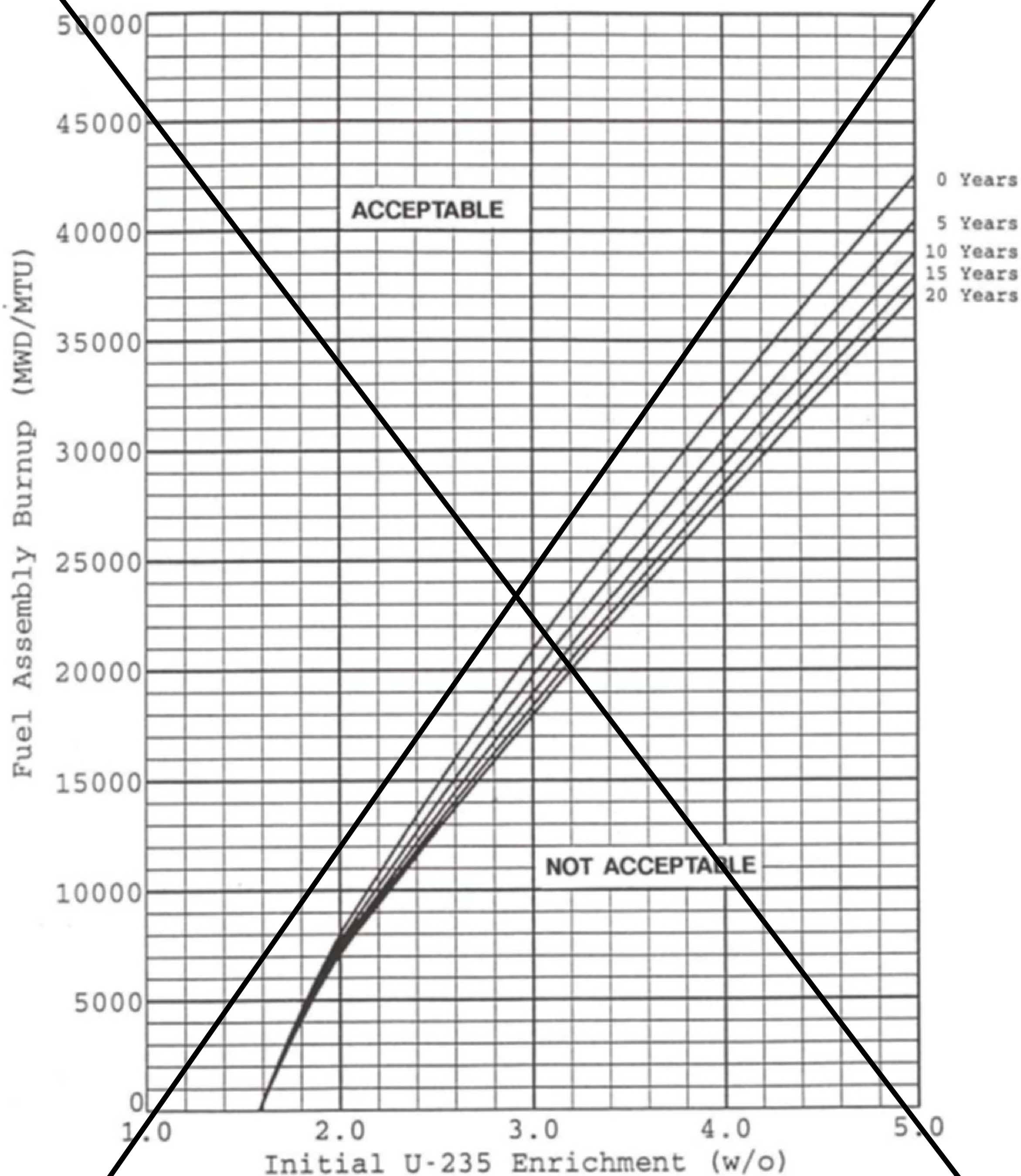

~~FIGURE 4.3.1-5 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 4 GAD~~

**Figure deleted**



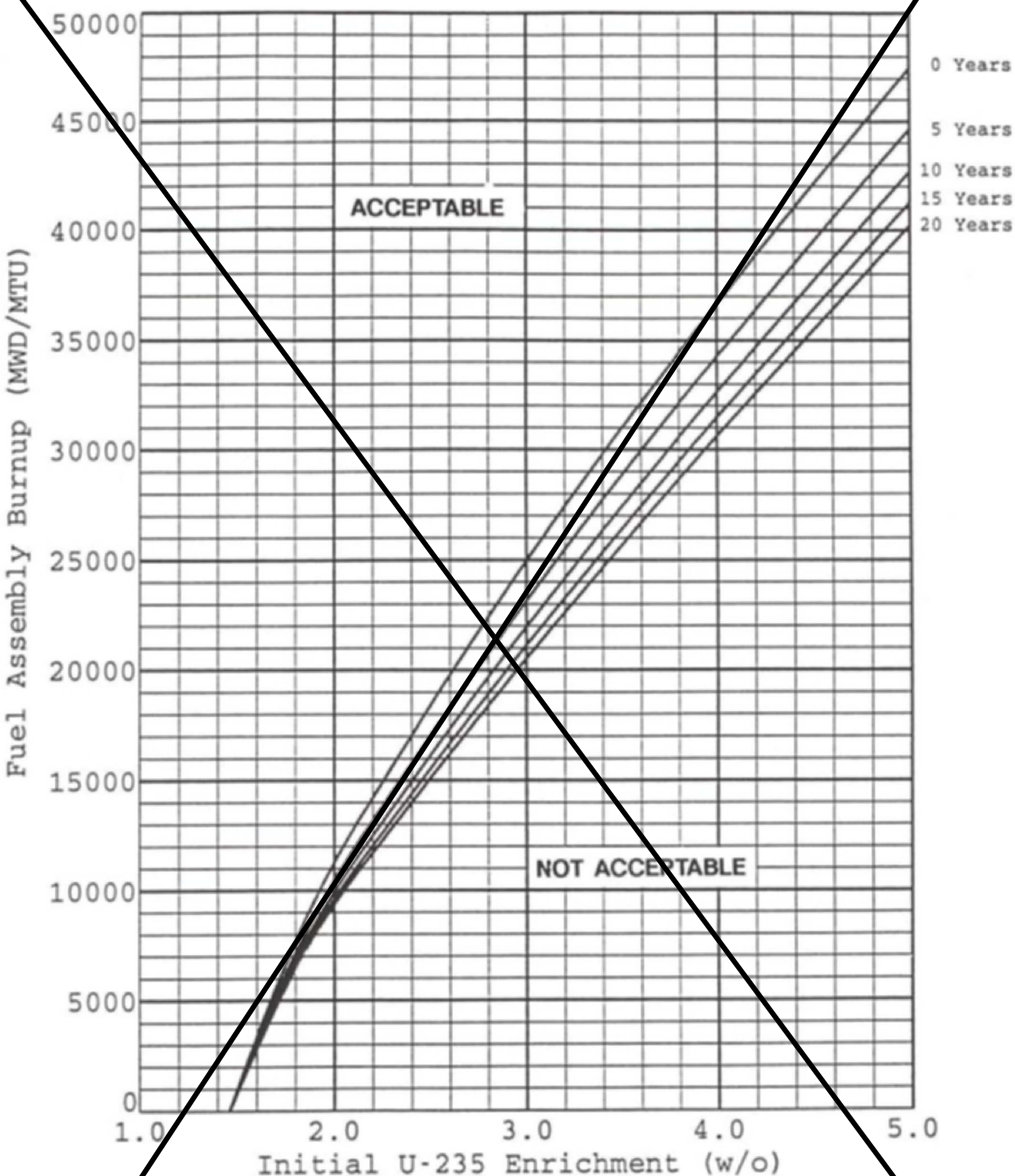

~~FIGURE 4.3.1-6 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 4 GAD~~

**Figure Deleted**



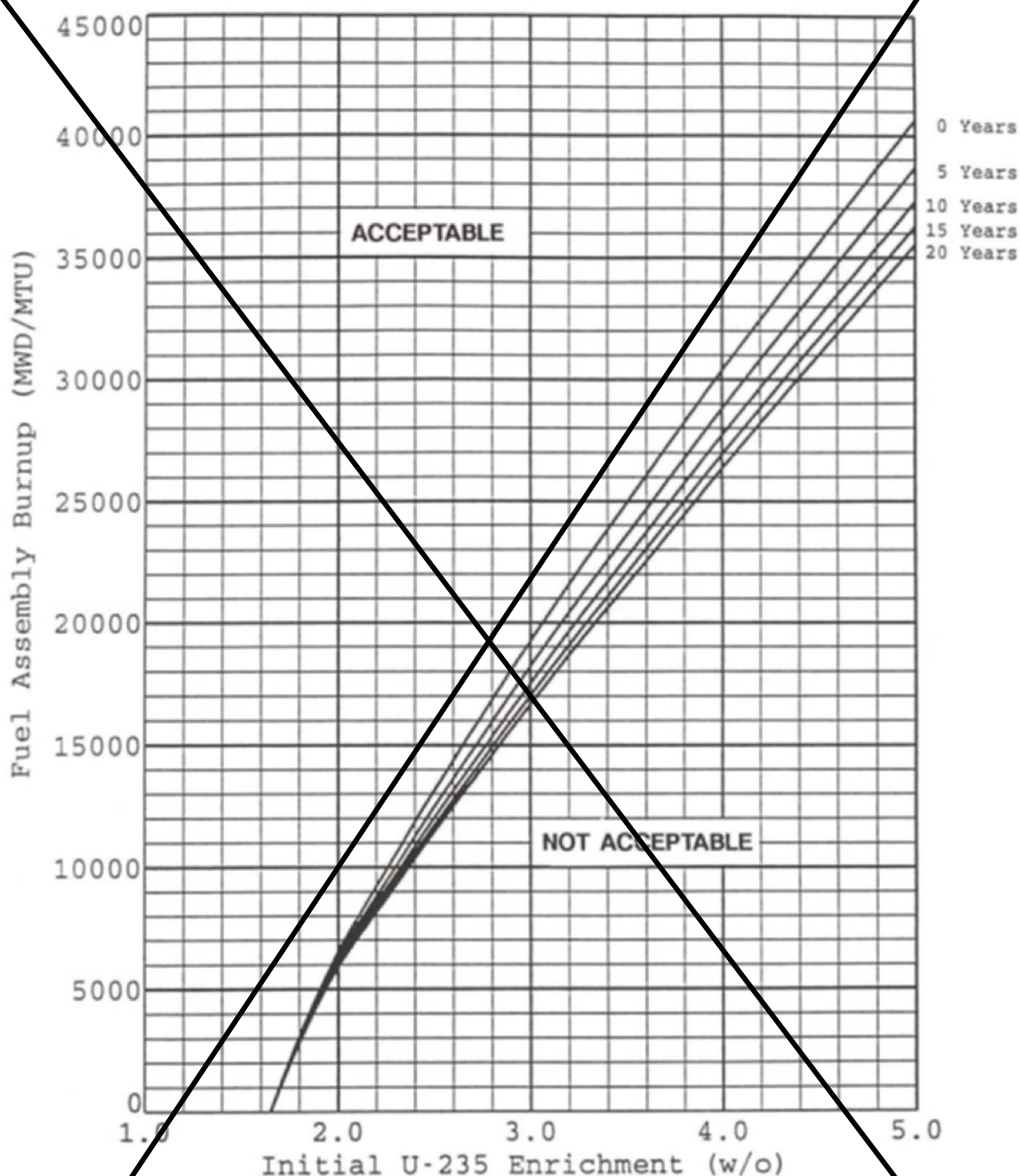

~~FIGURE 4.3.1-7 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 8 GAD~~

**Figure deleted**



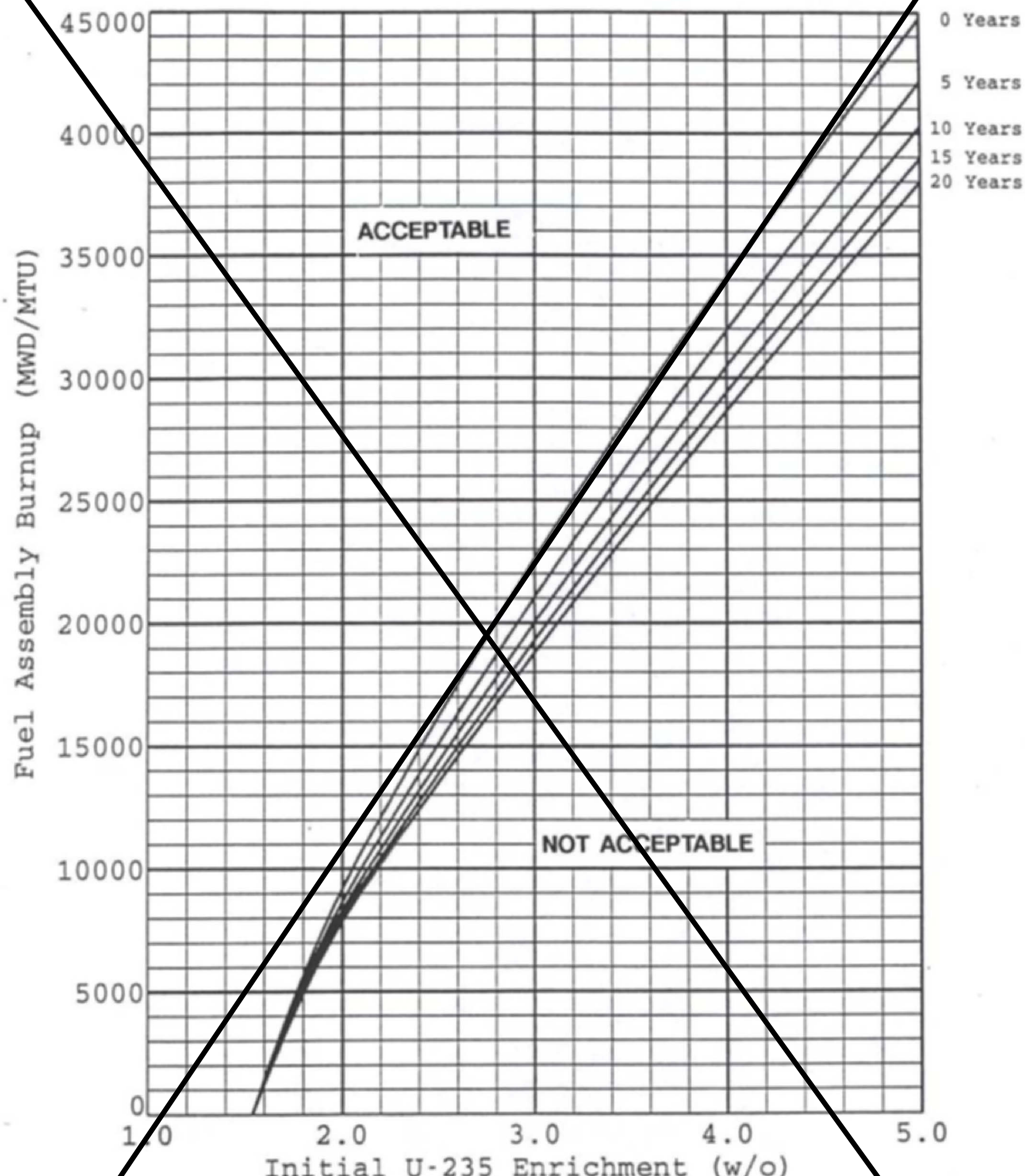

~~FIGURE 4.3.1-8 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 8 GAD~~

**Figure deleted**



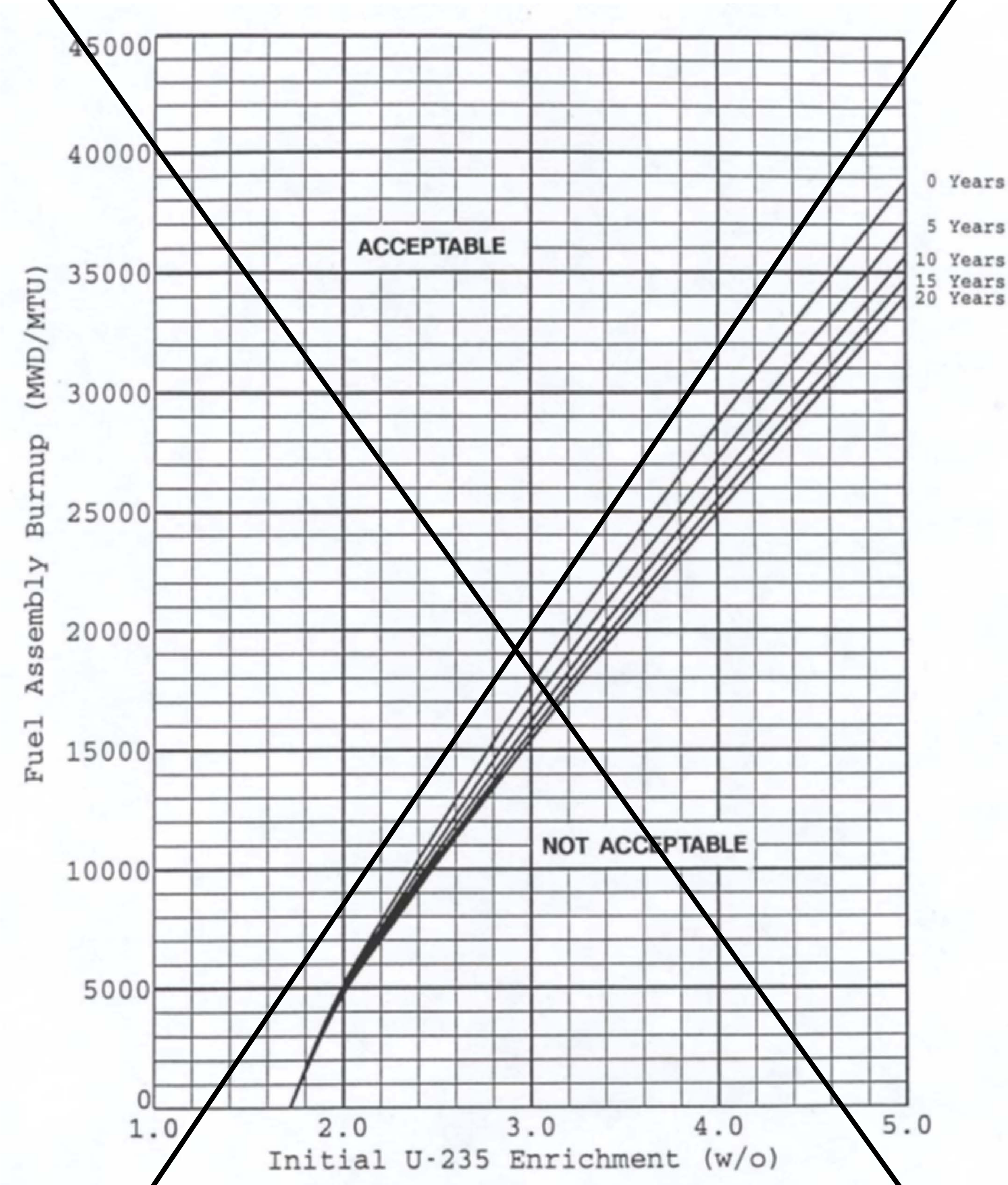

~~FIGURE 4.3.1-9 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 12 GAD~~

**Figure deleted**



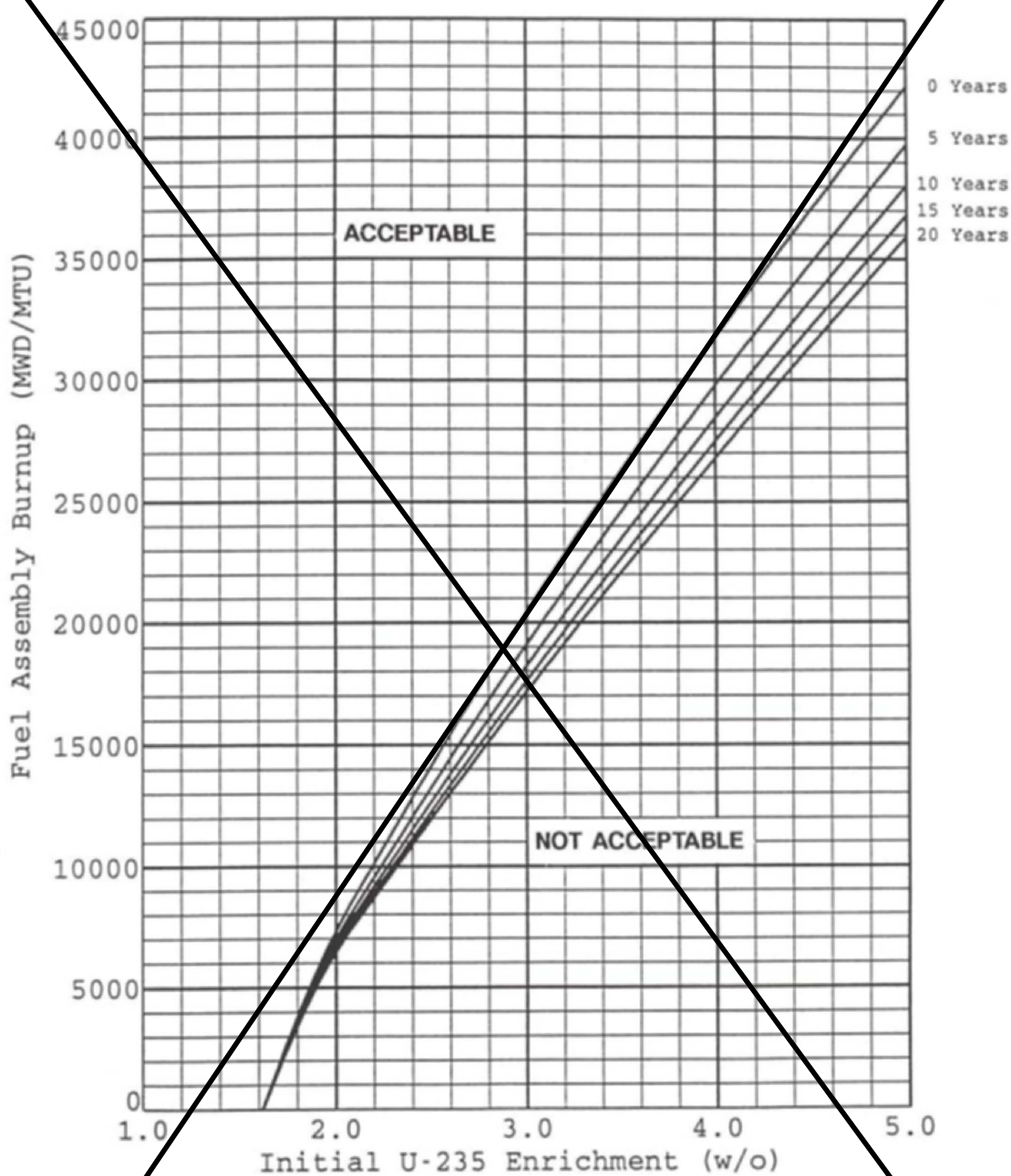

~~FIGURE 4.3.1-10 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 12 GAD~~

**Figure deleted**




~~FIGURE 4.3.1-11 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 16 or More GAD~~

**Figure deleted**




~~FIGURE 4.3.1-12 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 16 or More GAD~~

**Figure deleted**

# B 3.7 PLANT SYSTEMS

## B 3.7.17 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in the USAR (Ref. 1). These 2 compartments are referred to as Pool 1 and Pool 2. ~~Fuel stored in the Prairie Island fuel storage pools include fuel with the:~~

~~a. OFA designation, which includes the Westinghouse OFA and Vantage Plus designs; and~~

~~b. STD designation, which includes the Westinghouse Standard and Exxon fuel designs.~~

Criticality considerations provide the primary basis for storage limitations.

Pool 1 may contain up to 462 storage positions, except when the pool is used for cask laydown. In the latter case, only 266 storage positions are available since 4 storage racks must be removed to accommodate the storage cask. Pool 2 has up to 1120 storage positions.

Pools 1 and 2 are designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)), which have accumulated minimum burnups and decay times within the unrestricted domain according to ~~the applicable~~ Figure 3.7.17-1 ~~(OFA Fuel) or Figure 3.7.17-2 (STD Fuel),~~ in the accompanying LCO.

Fuel assemblies not meeting the criteria of ~~the applicable~~ Figure 3.7.17-1 ~~or Figure 3.7.17-2~~ shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

BASES

BACKGROUND (continued)

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting $k_{eff}$ of 1.00 be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO and maintaining boron concentration in accordance with LCO 3.7.16.

APPLICABLE SAFETY ANALYSES

The hypothetical criticality accidents can only take place during or as a result of the movement of an assembly (Ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Fuel Storage Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by verifying the appropriate checkerboarding after each fuel handling campaign, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for criticality accidents, the operation may be under the auspices of the accompanying LCO.

The spent fuel storage racks have been analyzed in accordance with the methodology contained in Reference 4~~5~~. That methodology ensures that the spent fuel rack multiplication factor, $k_{eff}$, is less than 0.95 as recommended by ANSI 57.2-1983 (Ref. 6) and NRC guidance (Ref. 3). The codes, methods and techniques contained in the methodology are used to satisfy this criterion on $k_{eff}$. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum

BASES

APPLICABLE SAFETY ANALYSES (continued)

of 5.0 **(nominal 4.95% + 0.05%)** weight percent U-235 while maintaining $k_{eff} \leq 0.95$ including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool ($k_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing gadolinium burnable poison.

The criticality analysis (Ref. **4**~~7~~) utilized the following storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment:

a. The first storage configuration utilizes a checkerboard loading pattern to accommodate new or low burnup fuel with a maximum enrichment of 5.0 w/o U-235. This configuration stores "burned" and "fresh" fuel assemblies in a 3x3 checkerboard pattern as shown in Figure 4.3.1-1. Fuel assemblies stored in "burned" cell locations are selected based on a combination of ~~fuel assembly type,~~ initial enrichment, discharge burnup and decay time (Figures 4.3.1-3 **and**~~through~~ 4.3.1-**4**~~12~~). The criteria for the fuel stored in the "burned" locations is also dependent on the **presence**~~number~~ of rods containing gadolinium in the center "fresh" fuel assembly. The use of empty cells is also an acceptable option for the **"fresh" and** "burned" cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.

Fuel assemblies that fall into the restricted range of Figure~~s~~ 3.7.17-1 ~~or 3.7.17-2~~ are required to be stored in "fresh" cell locations as shown in Figure 4.3.1-1. The criteria included in Figure~~s~~ 3.7.17-1 ~~and 3.7.17-2~~ for the selection of fuel assemblies to be stored in the "fresh" cell locations is based on a combination of ~~fuel assembly type,~~ initial enrichment, decay time and discharge burnup.

APPLICABLE SAFETY ANALYSES (continued)

b. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figure~~s~~ 3.7.17-1 ~~or 3.7.17-2, as applicable,~~ can be stored anywhere in the region with no special placement restrictions.

The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:

a. Separated by a vacant row of cells; ~~-~~or

b. The interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure 4.3.1-2).

Specification 3.7.17 and Section 4.3 ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the spent fuel rack criticality analysis (Ref. 4~~7~~).

The spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, and the Exxon fuel assembly types in storage in the spent fuel pool. ~~The OFA designation on the figures in Specification 3.7.17 and Section 4.3 bound all of the Westinghouse OFA and Vantage Plus fuel assemblies at Prairie Island. The STD designation on the figures in Specification 3.7.17 and Section 4.3 bound all of the Westinghouse STD and Exxon fuel assemblies at Prairie Island.~~

**Accident conditions which could increase the $k_{eff}$ were evaluated including:**
~~Most accident conditions in the spent fuel pool will not result in an increase in~~ ~~$k_{eff}$ of the racks. Examples of those accident conditions~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~which will not result in an increase in $k_{eff}$ are~~:

a. A **new** fuel assembly drop on the top of the racks;

b. A **new** fuel assembly **misloaded** ~~drop~~ between rack modules ~~and wall (rack design precludes this condition)~~; ~~and~~

**c.** ~~c.~~ A **new fuel assembly misloaded into an incorrect storage rack location;** ~~drop or placement of a fuel assembly into the cask loading area of the small pool.~~

**d. Intramodule water gap reduction due to a seismic event; and**

**e. Spent fuel pool temperature greater than 150 °F.**

~~However, two accidents can be postulated which could increase reactivity. The first postulated accident would be a loss of the spent fuel pool cooling system and the second would be a misload of a fuel assembly into a cell for which the restrictions on location, enrichment, burnup, decay time or gadolinium credit are not satisfied.~~

For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the **464** ~~750~~ ppm required to maintain $k_{eff}$ less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Westinghouse **Electric Company LLC** ~~Commercial Nuclear Fuel Division~~ calculations (Ref. **4**~~7~~) were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by ~~either of~~ these postulated accidents and to

maintain $k_{eff}$ less than or equal to 0.95. It was found that a spent fuel pool boron concentration of **730**~~1300~~ ppm was adequate to mitigate these postulated criticality related accidents and to maintain $k_{eff}$ less than or equal to 0.95.

Specification 3.7.16 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by

BASES

APPLICABLE SAFETY ANALYSES (continued)

a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.7.16 is consistent with the boron concentration limit required for a spent fuel cask containing fuel.

Section 4.3 requires that the spent fuel rack $k_{eff}$ be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 $k_{eff}$ design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

When the requirements of Specification 3.7.17 are not met, immediate action must be taken to move any noncomplying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with ~~the applicable~~ Figure 3.7.17-1 ~~(OFA Fuel) or Figure 3.7.17-2 (STD Fuel),~~ in the accompanying LCO, ensure the $k_{eff}$ of the spent fuel storage pool will always remain < 0.95, with credit given for boron in the water.

Fuel assemblies not meeting the criteria of ~~the appropriate~~ Figure 3.7.17-1 ~~or Figure 3.7.17-2~~ shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

BASES (continued)

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APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

---

ACTIONS

A.1

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

When the configuration of fuel assemblies stored in ~~-~~the spent fuel storage pool is not in accordance with ~~the applicable~~ Figure 3.7.17-1 or ~~Figure 3.7.17-2, or~~ **Specification** ~~paragraph~~ 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with ~~the applicable~~ Figure 3.7.17-1 or ~~Figure 3.7.17-2 or~~ Specification 4.3.1.1.

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

---

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with ~~the applicable~~ Figure 3.7.17-1 ~~or Figure 3.7.17-2~~ in the accompanying LCO. For

---

fuel assemblies in the restricted range of ~~the applicable~~ Figure 3.7.17-1~~ or Figure 3.7.17-2~~, performance of this SR will ensure compliance with Specification 4.3.1.1.

The Frequency of this SR is prior to storing or moving a fuel assembly.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.17.2

This SR verifies that the fuel assemblies in the spent fuel storage racks are stored in accordance with the requirements of LCO 3.7.17 and Section 4.3.1.1.

The intent of this SR is to not require completion of the spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.

The Frequency of this SR requires performance within 7 days after the completion of any fuel handling campaign which involves:

a. The relocation of fuel assemblies within the spent fuel pool; ~~-~~or

b. The addition of fuel assemblies to the spent fuel pool.

The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

The 7 day allowance for completion of this SR provides adequate time for completion of the spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the

spent fuel pool. If a fuel assembly is misloaded during the fuel handling campaign, the minimum boron concentration required by LCO 3.7.16 will ensure that the spent fuel rack $k_{eff}$ remains within limits until the spent fuel inventory verification is performed.

BASES (continued)

REFERENCES

1. USAR, Section 10.2.

2. ANSI/ANS-8.1-1983.

3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.

4. **"Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis".** ~~"Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks",~~ Westinghouse **Electric Company calculation CN-WFE-03-40,** ~~Commercial Nuclear Fuel Division~~, **November 11, 2004** ~~February 1993~~.

5. **Not Used.** ~~WCAP-14416- NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996~~.

6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.

~~7. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.~~

Exhibit C

**Proposed Technical Specification and Bases Changes (retyped)**

Technical Specification Pages

3.7.17-1
3.7.17-2
3.7.17-3
4.0-2
4.0-4
4.0-5
4.0-6
4.0-7
4.0-8

Bases pages
(for information only)

B 3.7.17-1
B 3.7.17-2
B 3.7.17-3
B 3.7.17-4
B 3.7.17-5
B 3.7.17-6
B 3.7.17-7
B 3.7.17-8
B 3.7.17-9

## 3.7 PLANT SYSTEMS

### 3.7.17 Spent Fuel Pool Storage

LCO 3.7.17 The combination of initial enrichment, burnup and decay time of each fuel assembly stored in the spent fuel pool shall be within the Unrestricted Region of Figure 3.7.17-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 -------------NOTE------------- LCO 3.0.3 is not applicable. --------------------------------- <br><br> Initiate action to move the noncomplying fuel assembly to an acceptable location. | <br><br><br> Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.17.1 | Verify by administrative means the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.17-1 or Specification 4.3.1.1. | Prior to storing or moving the fuel assembly |
| SR 3.7.17.2 | Verify spent fuel pool inventory. | Within 7 days after completion of a spent fuel pool fuel handling campaign |

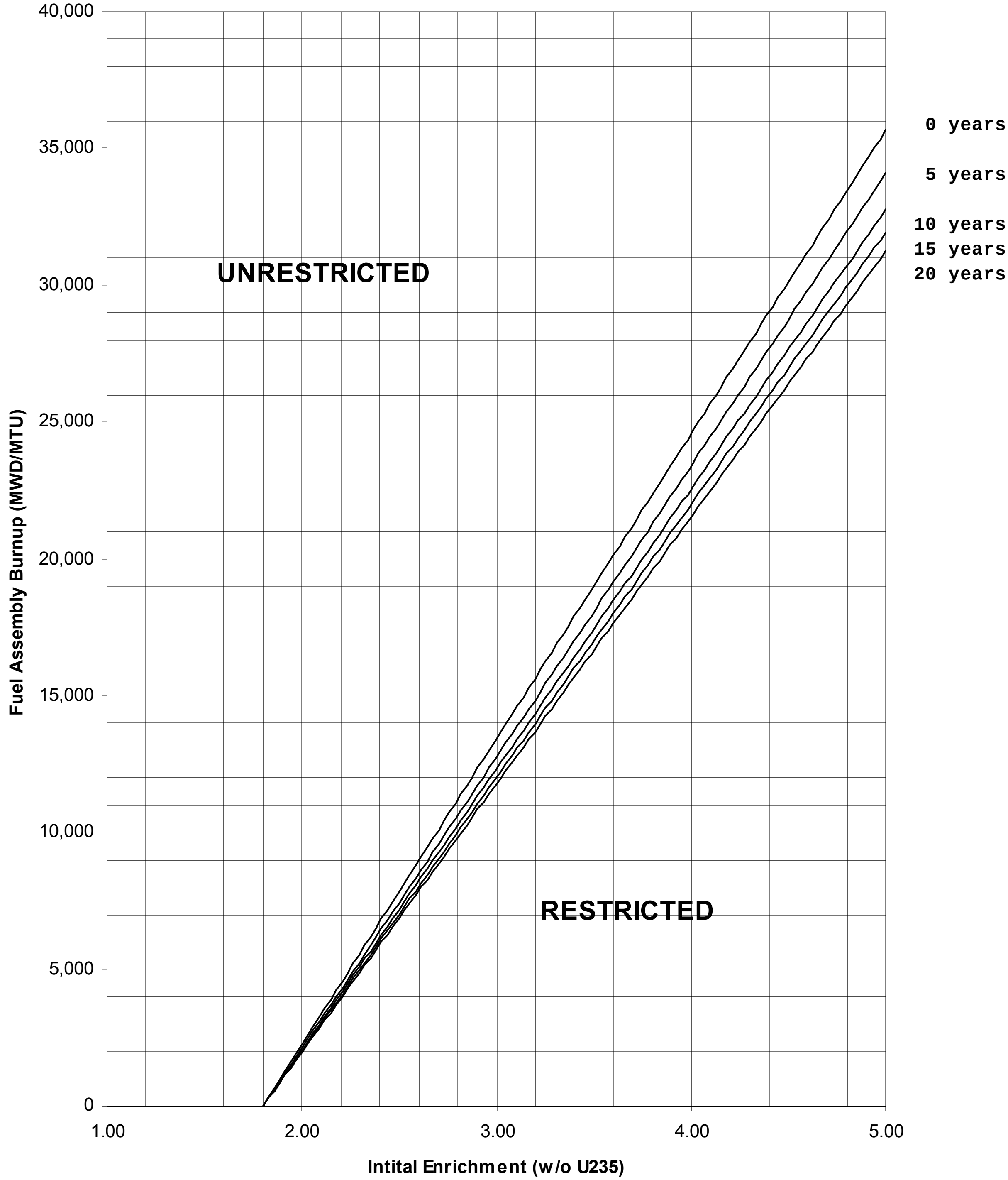



Figure 3.7.17-1
Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements

4.0 DESIGN FEATURES (continued)

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 1;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 730 ppm, which includes an allowance for uncertainties as described in Reference 1;

d. A nominal 9.5 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "unrestricted range" of Figure 3.7.17-1 may be allowed unrestricted storage in the fuel storage racks; and

f. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "restricted range" of Figure 3.7.17-1 will be stored in compliance with Figures 4.3.1-1 through 4.3.1-4.

## 4.0 DESIGN FEATURES

### 4.3 Fuel Storage (continued)

#### 4.3.3 Capacity

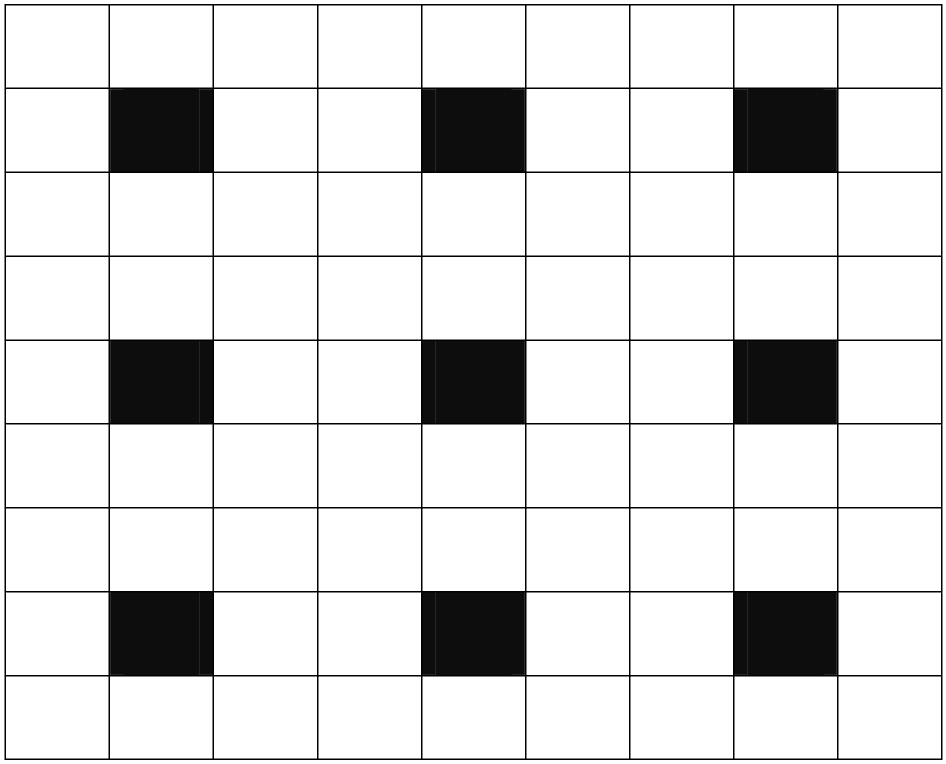
The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies not including those assemblies which can be returned to the reactor. The southeast corner of the small pool serves as the spent fuel cask lay down area. To facilitate plant evolutions, four additional storage racks, with a combined capacity of 196, may be temporarily installed in the cask lay down area to provide a total of 1582 storage locations (Ref. 3).

REFERENCES

1. "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis", Calculation Note Number CN-WFE-03-40, Westinghouse Electric Company, November 11, 2004.

2. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

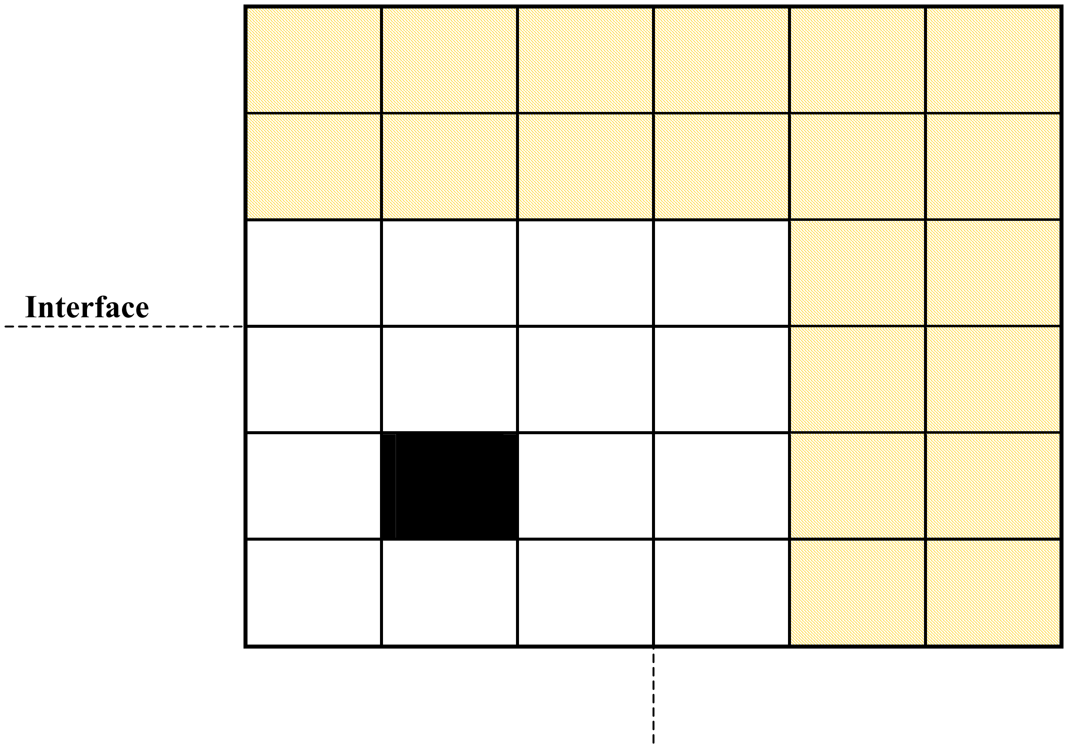
3. USAR, Section 10.2.



Fresh Fuel: Must be less than or equal to Nominal 4.95 w/o $^{235}U$
No restrictions on burnup

Burned Fuel: Must satisfy minimum burnup requirements of Figures 4.3.1-3 or 4.3.1-4 depending on presence of GAD rods in fresh fuel

Figure 4.3.1-1
Spent Fuel Pool Burned/Fresh Checkerboard Cell Layout




| | |
|---|---|
| Fresh Fuel: | Must be less than or equal to nominal 4.95 w/o $^{235}U$<br>No restrictions on burnup |
| Burned Fuel: | 3x3 Checkerboard Region<br>Must satisfy minimum burnup requirements of Figure 4.3.1-3 or 4.3.1-4 |
| Burned Fuel: | All Cell Unrestricted Region<br>Must satisfy minimum burnup requirements of Figure 3.7.17-1 |

Figure 4.3.1-2
Spent Fuel Pool Checkerboard Interface Requirements

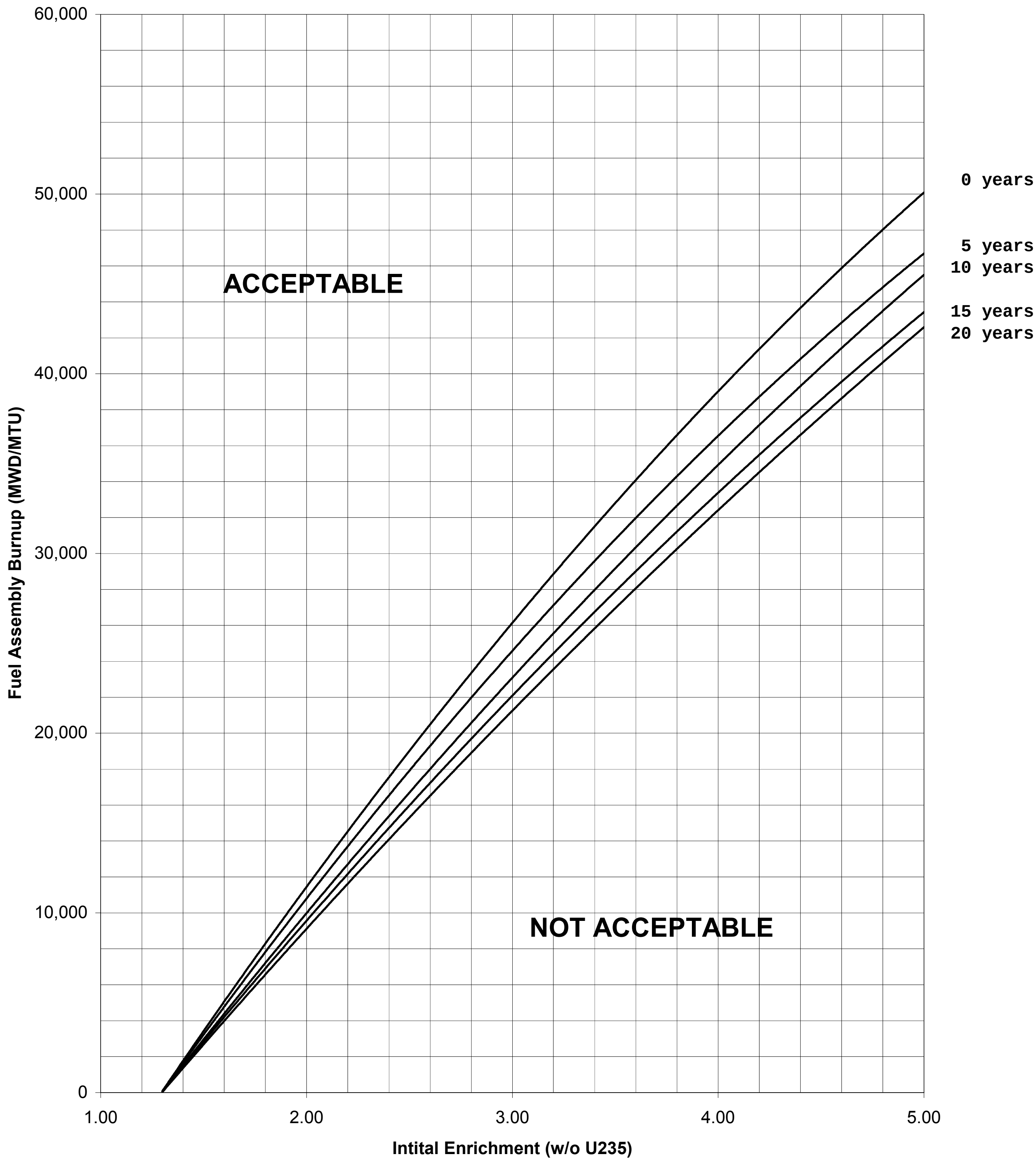



Figure 4.3.1-3
Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - No GAD

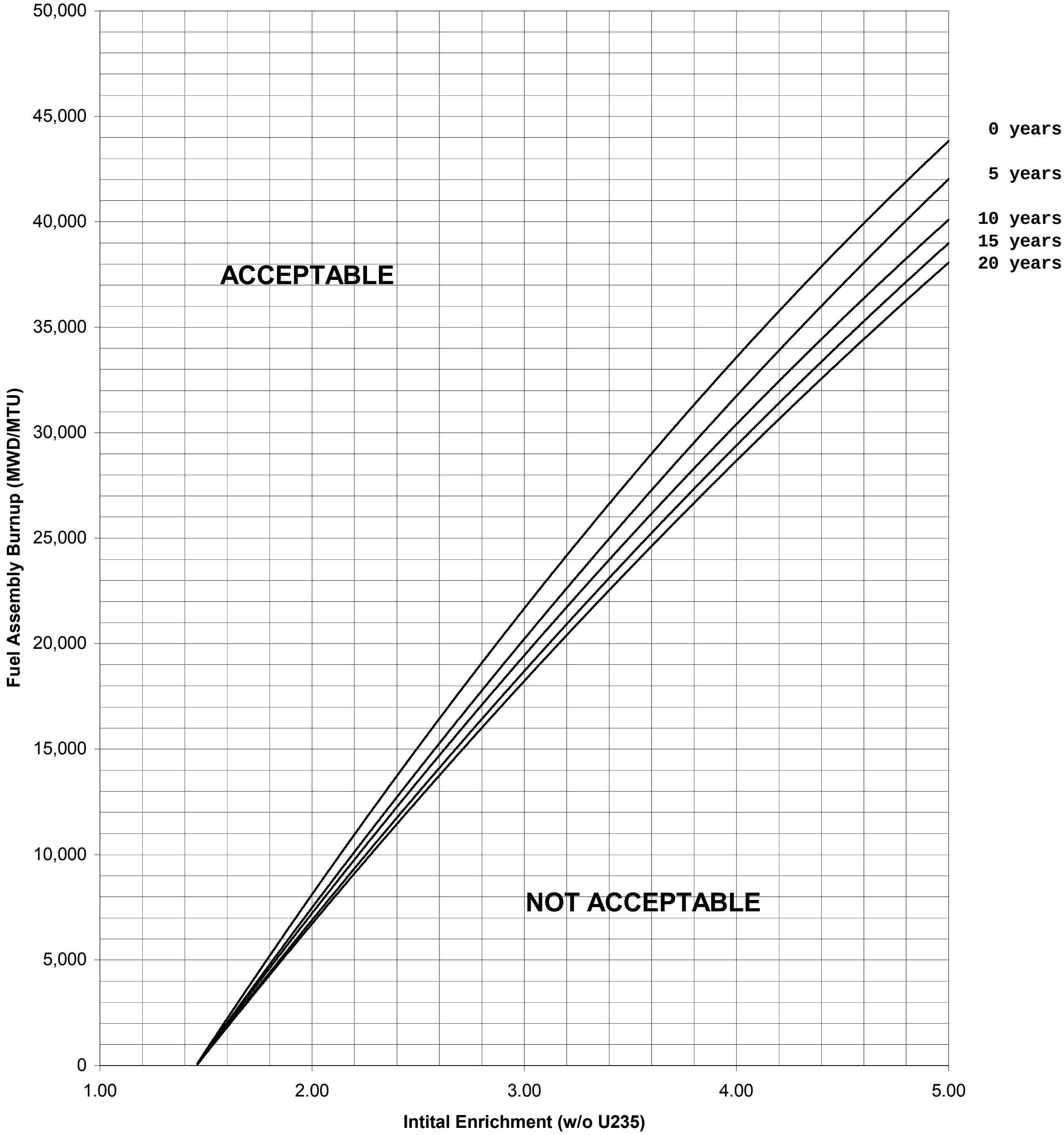



Figure 4.3.1-4
Spent Fuel Pool Checkerboard Region Burnup and Decay Time
Requirements - Fuel with GAD

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in the USAR (Ref. 1). These 2 compartments are referred to as Pool 1 and Pool 2.

Criticality considerations provide the primary basis for storage limitations.

Pool 1 may contain up to 462 storage positions, except when the pool is used for cask laydown. In the latter case, only 266 storage positions are available since 4 storage racks must be removed to accommodate the storage cask. Pool 2 has up to 1120 storage positions.

Pools 1 and 2 are designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)), which have accumulated minimum burnups and decay times within the unrestricted domain according to Figure 3.7.17-1 in the accompanying LCO.

Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

BASES

BACKGROUND (continued)

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting $k_{eff}$ of 1.00 be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO and maintaining boron concentration in accordance with LCO 3.7.16.

APPLICABLE SAFETY ANALYSES

The hypothetical criticality accidents can only take place during or as a result of the movement of an assembly (Ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, “Fuel Storage Pool Boron Concentration”) prevents criticality. By closely controlling the movement of each assembly and by verifying the appropriate checkerboarding after each fuel handling campaign, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for criticality accidents, the operation may be under the auspices of the accompanying LCO.

The spent fuel storage racks have been analyzed in accordance with the methodology contained in Reference 4. That methodology ensures that the spent fuel rack multiplication factor, $k_{eff}$, is less than 0.95 as recommended by ANSI 57.2-1983 (Ref. 6) and NRC guidance (Ref. 3). The codes, methods and techniques contained in the methodology are used to satisfy this criterion on $k_{eff}$. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum

BASES

APPLICABLE SAFETY ANALYSES (continued)

of 5.0 (nominal 4.95% $\pm$ 0.05%) weight percent U-235 while maintaining $k_{eff} \leq 0.95$ including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool ($k_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing gadolinium burnable poison.

The criticality analysis (Ref. 4) utilized the following storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment:

a. The first storage configuration utilizes a checkerboard loading pattern to accommodate new or low burnup fuel with a maximum enrichment of 5.0 w/o U-235. This configuration stores “burned” and “fresh” fuel assemblies in a 3x3 checkerboard pattern as shown in Figure 4.3.1-1. Fuel assemblies stored in “burned” cell locations are selected based on a combination of initial enrichment, discharge burnup and decay time (Figures 4.3.1-3 and 4.3.1-4). The criteria for the fuel stored in the “burned” locations is also dependent on the presence of rods containing gadolinium in the center “fresh” fuel assembly. The use of empty cells is also an acceptable option for the “fresh” and “burned” cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.

   Fuel assemblies that fall into the restricted range of Figure 3.7.17-1 are required to be stored in “fresh” cell locations as shown in Figure 4.3.1-1. The criteria included in Figure 3.7.17-1 for the selection of fuel assemblies to be stored in the “fresh” cell locations is based on a combination of initial enrichment, decay time and discharge burnup.

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figure 3.7.17-1 can be stored anywhere in the region with no special placement restrictions.

The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:

a. Separated by a vacant row of cells; or

b. The interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure 4.3.1-2).

Specification 3.7.17 and Section 4.3 ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the spent fuel rack criticality analysis (Ref. 4).

The spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, and the Exxon fuel assembly types in storage in the spent fuel pool.

Accident conditions which could increase the $k_{eff}$ were evaluated including:

a. A new fuel assembly drop on the top of the racks;

b. A new fuel assembly misloaded between rack modules;

c. A new fuel assembly misloaded into an incorrect storage rack location;

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. Intramodule water gap reduction due to a seismic event; and

e. Spent fuel pool temperature greater than 150°F.

For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 464 ppm required to maintain $k_{eff}$ less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Westinghouse Electric Company LLC calculations (Ref. 4) were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by these postulated accidents and to maintain $k_{eff}$ less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 730 ppm was adequate to mitigate these postulated criticality related accidents and to maintain $k_{eff}$ less than or equal to 0.95.

Specification 3.7.16 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.7.16 is consistent with the boron concentration limit required for a spent fuel cask containing fuel.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Section 4.3 requires that the spent fuel rack $k_{eff}$ be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 $k_{eff}$ design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

When the requirements of Specification 3.7.17 are not met, immediate action must be taken to move any noncomplying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.17-1 in the accompanying LCO, ensure the $k_{eff}$ of the spent fuel storage pool will always remain < 0.95, with credit given for boron in the water.

Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

BASES (continued)

ACTIONS

A.1

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

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with Figure 3.7.17-1 or Specification 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.17-1 or Specification 4.3.1.1.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 in the accompanying LCO. For fuel assemblies in the restricted range of Figure 3.7.17-1 performance of this SR will ensure compliance with Specification 4.3.1.1.

The Frequency of this SR is prior to storing or moving a fuel assembly.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.17.2

This SR verifies that the fuel assemblies in the spent fuel storage racks are stored in accordance with the requirements of LCO 3.7.17 and Section 4.3.1.1.

The intent of this SR is to not require completion of the spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.

The Frequency of this SR requires performance within 7 days after the completion of any fuel handling campaign which involves:

a. The relocation of fuel assemblies within the spent fuel pool; or

b. The addition of fuel assemblies to the spent fuel pool.

The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

The 7 day allowance for completion of this SR provides adequate time for completion of the spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the spent fuel pool. If a fuel assembly is misloaded during the fuel handling campaign, the minimum boron concentration required by LCO 3.7.16 will ensure that the spent fuel rack $k_{eff}$ remains within limits until the spent fuel inventory verification is performed.

BASES (continued)

REFERENCES

1. USAR, Section 10.2.

2. ANSI/ANS-8.1-1983.

3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.

4. "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis", Westinghouse Electric Company calculation CN-WFE-03-40, November 11, 2004.

5. Not Used.

6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.

Exhibit D

# Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis

87 pages follow

WESTINGHOUSE ELECTRIC COMPANY LLC

| Calculation Note Number | | Revision | SAP Charge Number | Page |
|---|---|---|---|---|
| CN-WFE-03-40 | | 0 | 603137 | 1 |
| Project | Releasable (Y/N) | Shop Order Number | Files Attached (Y/N) | Total No. Pages |
| Prairie Island Units 1 & 2 | Y | N/A | Y | 87 |

# Title: Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis

| Author(s) Name(s) | Signature / Date | Scope |
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| Verifier(s) Name(s) | Signature / Date | Scope |
|---|---|---|
| E. Fuentes | 11/11/04 | Qualified Verifier |

| Manager Name | Signature / Date |
|---|---|
| Greg Vincent | 11/11/04 |



Westinghouse

## Record of Revisions

| Rev | Date* | Revision Description |
|---|---|---|
| 0 | | Original Issue |

* Final Approval Date. For current revision, date is either on the cover page or on the EDMS approval sheet when using electronic approval.

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# 1.0 Objective

This report presents the results of criticality analyses for the Prairie Island Units 1 & 2 spent fuel storage racks with credit for assembly burnup, Fuel Burnable Absorber ($Gd_2O_3$), $^{241}$Pu decay and soluble boron. The primary objectives of this calculation are as follows:

1. To determine the design basis fuel assembly for all of the fuel assembly storage configurations. They include the "All-Cell" and "3x3 array" fuel assembly storage configurations.

2. To determine the assembly burnup versus initial enrichment limits required for safe storage of fuel assemblies in the "All-Cell" storage configuration

3. To determine the assembly burnup versus initial enrichment limits required for safe storage of peripheral fuel assemblies in the "3x3 array" with the center fuel assembly initially enriched to 4.95 w/o $^{235}$U. This will be accomplished with credit for 5, 10, 15, and 20 years of $^{241}$Pu decay.

4. To determine the assembly burnup versus initial enrichment limits required for safe storage of peripheral fuel assemblies in the "3x3 array" with the center fuel assembly initially enriched to 4.95 w/o $^{235}$U and shimmed with 4 $Gd_2O_3$ rods. These limits will be derived based upon a $Gd_2O_3$ concentration of 4.0 w/o. This will be accomplished with credit for 5, 10, 15, and 20 years of $^{241}$Pu decay.

5. To determine if the current interface between storage configurations is still valid

6. To determine the amount of soluble boron required to maintain $k_{eff}$ less than or equal to 0.95, including all biases and uncertainties, assuming the most limiting plausible reactivity accident.

The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 2 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 3. Reference 2 was reviewed and approved by the US NRC. The methodology employed in this analysis and in Reference 2 employs axially distributed burnups to represent discharged fuel assemblies. This calculation note was prepared according to Westinghouse Procedure EP-302 (Reference 1).

## 1.1 Design Criteria

The design criteria are consistent with Reference 4 and NRC guidance given in Reference 5. Section 1.4 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria follows.

1. Determine the fresh and spent fuel storage configurations using no soluble boron conditions such that the 95/95 upper tolerance limit value of $k_{eff}$, including applicable biases and uncertainties, is less than 0.999. This is accomplished with infinite arrays of either fresh or spent fuel assembly configurations. Note that the actual NRC $k_{eff}$ limit for this condition is unity. Therefore, an additional safety margin equal to 0.001 $\Delta k_{eff}$ units is included in the infinite array analysis results. Additional margin to the $k_{eff}$ limit will be identified based upon the KENO results for the entire spent fuel pool #2.

2. Determine the amount (ppm) of soluble boron necessary to reduce the $k_{eff}$ value of all storage configurations by at least 0.05 $\Delta k_{eff}$ units. This is accomplished by constructing a KENO model for spent fuel pool #2 which includes the storage configurations which are least sensitive to changes in soluble boron concentration. As an example, storage configurations which contain depleted fuel assemblies (and represented by depleted isotopics) are less reactivity-sensitive to changes in soluble boron concentration than an assembly represented by zero burnup and a relatively low initial fuel enrichment. Note that spent fuel pool #2 is much larger than spent fuel pool #1 and therefore the results will be bounding for both spent fuel pools.

3. Determine the amount of soluble boron necessary to compensate for 5% of the maximum burnup credited in any storage configuration. In addition, determine the amount of soluble boron necessary to account for a reactivity depletion uncertainty equal to 1.0% $\Delta k_{eff}$ per 30,000 MWD/MTU of credited assembly burnup. This is accomplished by multiplying this derivative by the maximum burnup credited in any storage configuration and converting to soluble boron using the data generated in Step 2.

4. Determine the largest increase in reactivity caused by postulated accidents and the corresponding amount of soluble boron needed to offset this reactivity increase.

An alternative form of expressing the soluble boron requirements is given in Reference 3. The final soluble boron requirement is determined from the following summation.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

Where:

$SBC_{TOTAL}$ = total soluble boron credit requirement (ppm).

$SBC_{95/95}$ = soluble boron requirement for 95/95 $k_{eff}$ less than or equal to 0.95 (ppm).

$SBC_{RE}$ = soluble boron required to account for burnup and reactivity depletion uncertainties (ppm).

$SBC_{PA}$ = soluble boron required to maintain $k_{eff}$ less than or equal to 0.95 under accident conditions (ppm).

For purposes of the analyses contained herein, minimum burnup limits established for fuel assemblies to be stored in the storage racks do include burnup credit established in a manner which takes into account approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses. Also, the axial reactivity effect associated with the absence of $Gd_2O_3$ at both ends of the fuel assembly was directly included in this analysis

## 1.2 Design Approach

The Soluble Boron Credit Methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and, for example, to eliminate the need to credit any of the degraded Boraflex. Boraflex in the spent fuel racks is not credited in this analysis.

All of the storage cells modeled in this analysis employ a realistic representation of the pitch between storage locations. The square storage cell pitch for the "All-Cell" and "3x3" fuel assembly storage configurations employed for this analysis is equal to 9.5 inches.

The selection of the design basis fuel assembly type was based on an evaluation of the variety of fuel assemblies employed in the reactor to date and selecting the most reactive type for a given fuel assembly storage configuration. The candidate fuel assembly types include the Westinghouse and Exxon 14x14 Standard (STD), the Westinghouse 14x14 Optimized (OFA), and the Exxon TOPROD fuel assembly designs. The Westinghouse 14x14 OFA fuel assembly has been evaluated to be the design basis fuel assembly to represent fresh fuel assemblies in the center location of the "3x3" fuel assembly storage configurations. The Westinghouse 14x14 Standard fuel assembly has been evaluated to be the design basis fuel assembly to represent discharged fuel assemblies in the "All-Cell", and peripheral locations of the "3x3" fuel assembly storage configurations. The most reactive moderator conditions (water density equal to 1.0 g/cc) will be employed for each fuel assembly storage configuration such that the analysis results are valid over the nominal spent fuel temperature range (50 to 150 degrees Fahrenheit).

The reactivity characteristics of the storage racks were evaluated using infinite lattice analyses; this environment was employed in the evaluation of the burnup limits versus initial enrichment as well as the evaluation of physical tolerances and uncertainties. A full spent fuel pool model was also employed to evaluate soluble boron worth, the reactivity worth of postulated accidents, and the multiplication factor for the zero soluble boron condition.

## 1.3 Methodology

This section describes the methodology employed to assure the criticality safety of the spent fuel pools and to define limits placed on fresh and depleted fuel assembly storage configurations. The analysis methodology employs: (1) SCALE-PC, a personal computer version of the SCALE-4.3 code system, as documented in Reference 6, with the updated SCALE-4.3 version of the 44 group ENDF/B-V neutron cross section library, and (2) the two-dimensional integral transport code DIT, Reference 7, with an ENDF/B-VI neutron cross section library.

SCALE-PC was used for calculations involving infinite arrays for the “All-Cell” and “3x3” fuel assembly storage configurations. In addition, it was employed in a full representation of spent fuel pool #2 to evaluate soluble boron worth and postulated accidents.

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. All references to KENO in the text to follow should be interpreted as referring to the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

### 1.3.1 SCALE-PC

The SCALE system was developed for the Nuclear Regulatory Commission to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on specific classes of personal computers.

### 1.3.2 Validation of SCALE-PC

Validation of SCALE-PC for purposes of fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs. The first program is the Babcock & Wilcox (B&W) experiments carried out in support of Close Proximity Storage of Power Reactor Fuel, Reference 8. The second program is the Pacific Northwest Laboratory (PNL) Program carried out in support of the design of Fuel Shipping and Storage Configurations; the experiments of current interest to this effort are documented in Reference 9. Reference 10, as well as several of the relevant thermal experiment evaluations in Reference 11, were found to be useful in updating pertinent experimental data for the PNL experiments.

Nineteen experimental configurations were selected from the B&W experimental program; these consisted of the following experimental cores: Core X, the seven measured configurations of Core XI, Cores XII through XXI, and Core XIIIA. These analyses employed measured critical data, rather than the extrapolated configurations to a

fixed critical water height reported in Reference 8, so as to avoid introducing possible biases or added uncertainties associated with the extrapolation techniques. In addition to the active fuel region of the core, the full environment of the latter region, including the dry fuel above the critical water height, was represented explicitly in the analyses.

The B&W group of experimental configurations employed variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or Borated Aluminum plates of different boron contents in the water channels between rod clusters were measured. Table 1-1 summarizes the results of these analyses.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blackness. As in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 1-2 summarizes the results of these analyses.

The approach employed for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 12. For a given KENO calculated value of $k_{eff}$ and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed by the following equation; by this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

$$k_{95/95} = k_{KENO} + \Delta k_{bias} + M_{95/95}\left(\sigma_m^2 + \sigma_{KENO}^2\right)^{1/2}$$

Where,

$k_{keno}$ is the KENO calculated multiplication factor,

$\Delta k_{bias}$ is the mean calculational method bias,

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses, and is obtained from the Tables of Reference 13.

$\sigma_m^2$ is the mean calculational method variance deduced from the validation analyses,

$\sigma_{KENO}^2$ is the square of the KENO standard deviation.

$M_{95/95}(\sigma_m^2+\sigma_{KENO}^2)^{1/2}$ is equal to the methodology uncertainty

The equation for the mean calculational methods bias is as follows.

$$\Delta k_{bias} = \frac{1}{n}\sum_{i=1}^{n}\left(1-k_i\right)$$

Where

$k_i$ is the $i^{th}$ value of the multiplication factor for the validation lattices of interest.

The equation for the mean calculational variance of the relevant validating multiplication factors is as follows.

$$\sigma_m^2 = \frac{n\sum_1^n \left(k_i - k_{ave}\right)^2 \sigma_i^2}{\left(n-1\right)\sum_1^n \frac{1}{\sigma_i^2}} - \sigma_{ave}^2$$

where $k_{ave}$ is given by the following equation.

$$k_{ave} = \frac{\sum_1^n \frac{k_i}{\sigma_i^2}}{\sum_1^n \frac{1}{\sigma_i^2}},$$

$\sigma_{ave}^2$ is given by the following equation.

$$\sigma_{ave}^2 = \frac{\sum_1^n \sigma_i^2 G_i}{\sum_1^n G_i},$$

$G_i$ is the number of generations.

For purposes of this bias evaluation, the data points of Table 1-1 and Table 1-2 are pooled into a single group. With this approach, the mean calculational methods bias, $\Delta k_{bias}$, and the mean calculational variance, ($\sigma_m{}^{2)}$, calculated by equations given above, are determined to be 0.00259 and $(0.002882)^2$, respectively. The magnitude of $M_{95/95}$ is obtained from Reference 13 for the total number of pooled data points, 30.

The magnitude of $k_{95/95}$ is given by the following equation for SCALE 4.3 KENO analyses employing the 44-group ENDF/B-V neutron cross section library and for analyses where these experiments are a suitable basis for assessing the methods bias and calculational variance.

$$k_{95/95} = k_{KENO} + 0.00259 + 2.22\left[(0.00288)^2 + \sigma_{KENO}^2\right]^{\frac{1}{2}}$$

Based on the above analyses, the mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier are deduced as 0.00259, $(0.00288)^2$, and 2.22, respectively

### 1.3.3 Application to Fuel Storage Pool Calculations

As noted above, the CSAS25 control module was employed to execute the functional modules within SCALE-PC. The CSAS25 control module was used to analyze either infinite arrays of single or multiple storage cells or the full spent storage pool.

Standard material compositions were employed in the SCALE-PC analyses consistent with the design input given in Section 2.0; these data are listed in Table 1-3. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS25 module using input consistent with the data of Table 1-3. For burned fuel representations, the fuel isotopics were derived from the DIT code as described below.

### 1.3.4 The DIT Code

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells through the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI). Cross sections have been collapsed into an 89-group structure which is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually 4 broad groups), for use in the assembly flux calculation. A B1 assembly leakage correction is performed to modify the spectrum according to the assembly in- or out-leakage. Following the flux calculation, a depletion step is performed to generate a set of region-wise isotopic concentrations at the end of a burnup interval. An extensive set of depletion chains are available, containing 33 actinide nuclides in the thorium, uranium and plutonium chains, 171 fission products, the gadolinium, erbium and boron depletable absorbers, and all structural nuclides. The spectrum-depletion sequence of calculations is repeated over the life of the fuel assembly. Several restart capabilities provide the temperature, density, and boron concentration dependencies needed for three-dimensional calculations with full thermal-hydraulic feedback effects.

The DIT code and its cross Section library are employed in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For the purpose of spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each complete set of fuel isotopics is reduced to a smaller set of burned fuel isotopics at specified time points after discharge. The latter burned fuel representation includes the following nuclides: $^{235}$U, $^{236}$U, $^{238}$U, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{149}$Sm, $^{16}$O, and $^{10}$B. The DIT code lists the Samarium-149 isotopics for $^{149}$Sm and $^{149D}$Sm (a metastable isomer). Since $^{149}$Sm is a stable isotope, the concentration of this Samarium isotope is the sum of the individual concentration of these two isomers.

The isotopic number densities from the DIT calculation are based upon pin cell averaged values. The input to KENO calculations requires that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of $^{10}$B supplied to KENO is such that the KENO and DIT assembly $k_\infty$ values (at room temperature and unborated conditions) agree to within one sigma of the KENO calculation.

## 1.4 Assumptions

- The Westinghouse OFA was modeled as the design basis fuel assembly to conservatively represent all fuel assemblies residing in the center locations of the “3x3” fuel assembly storage configurations.

- The Westinghouse Standard fuel assembly was modeled as the design basis fuel assembly to conservatively represent all fuel assemblies residing in the “All-Cell” and peripheral locations of the “3x3” fuel assembly storage configurations.

- Fresh Standard and OFA fuel assemblies were conservatively modeled with a $UO_2$ density equal to 10.576 g/cc (96.5% of theoretical density). This translates into a pellet density equal 97.6% of theoretical density with a 1.1% dishing (void) fraction.

- All fuel assemblies, fresh and depleted, were conservatively modeled as containing solid right cylindrical pellets and uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs which incorporate lower enrichment blanket or annular pellets.

- All of the Boraflex poison material residing in the storage racks is conservatively omitted for this analysis .

- The intra module water gaps were conservatively modeled as 1.0 inches.

**Table 1-1 Calculational Results for Cores X Through XXI of the B&W Close Proximity Experiments**

| Core | Run No. | KENO $k_{eff}$ | Plate Type[1] | Spacing[2] |
|---|---|---|---|---|
| X | 2348 | 0.99610 ± 0.00084 | none | 3 |
| XI | 2355 | 1.00049 ± 0.00080 | SS-304 | 1 |
| XI | 2359 | 0.99884 ± 0.00077 | SS-304 | 1 |
| XI | 2360 | 1.00315 ± 0.00081 | SS-304 | 1 |
| XI | 2361 | 0.99831 ± 0.00080 | SS-304 | 1 |
| XI | 2362 | 1.00060 ± 0.00078 | SS-304 | 1 |
| XI | 2363 | 0.99957 ± 0.00078 | SS-304 | 1 |
| XI | 2364 | 1.00246 ± 0.00080 | SS-304 | 1 |
| XII | 2370 | 0.99990 ± 0.00082 | SS-304 | 2 |
| XIII | 2378 | 0.99754 ± 0.00089 | B/Al | 1 |
| XIIIA | 2423 | 0.99575 ± 0.00087 | B/Al | 1 |
| XIV | 2384 | 0.99465 ± 0.00086 | B/Al | 1 |
| XV | 2388 | 0.99158 ± 0.00084 | B/Al | 1 |
| XVI | 2396 | 0.99230 ± 0.00088 | B/Al | 2 |
| XVII | 2402 | 0.99478 ± 0.00079 | B/Al | 1 |
| XVIII | 2407 | 0.99440 ± 0.00083 | B/Al | 2 |
| XIX | 2411 | 0.99821 ± 0.00081 | B/Al | 1 |
| XX | 2414 | 0.99498 ± 0.00082 | B/Al | 2 |
| XXI | 2420 | 0.99318 ± 0.00094 | B/Al | 3 |

[1] Entry indicates metal separating unit assemblies.
[2] Entry indicates spacing between unit assemblies in units of fuel rod pitch.

**Table 1-2. Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping and Storage Configurations**

| Experiment | $k_{eff}$ | Comments |
|---|---|---|
| | | |
| 043 | 0.99787 ± 0.00106 | Uniform rectangular array, no poison |
| 044 | 1.00104 ± 0.00102 | “ |
| 045 | 0.99955 ± 0.00101 | “ |
| 046 | 0.99960 ± 0.00103 | “ |
| 061 | 0.99792 ± 0.00099 | 2 x 2 array of rod clusters, no poison |
| 062 | 0.99628 ± 0.00096 | “ |
| 064 | 0.99696 ± 0.00103 | 2 x 2 array of rod clusters, 0.302 cm thick SS-304 cross |
| 071 | 0.99970 ± 0.00101 | 2 x 2 array of rod clusters, 0.485 cm thick SS-304 cross |
| 079 | 0.99463 ± 0.00102 | 2 x 2 array of rod clusters, cross of 0.3666 g boron/cm$^2$ |
| 087 | 0.99423 ± 0.00099 | 2 x 2 array of rod clusters, cross of 0.1639 g boron/cm$^2$ |
| 093 | 0.99787 ± 0.00098 | 2 x 2 array of rod clusters, cross of 0.1425 g boron/cm$^2$ |

**Table 1-3. Standard Material Compositions Employed in Criticality Analysis for Prairie Island Units 1 & 2 Spent Fuel Storage Racks**

| Material | Element | Weight Fraction |
|---|---|---|
| Zircaloy-4, Density = 6.56 g/cm$^3$ @ 293.15 K | Zr | 0.9829 |
| | Sn | 0.0140 |
| | Fe | 0.0021 |
| | Cr | 0.0010 |
| 4.0 w/o $Gd_2O_3$ @ 293.15 K | $^{154}Gd$ | 0.0007304 |
| | $^{155}Gd$ | 0.0050360 |
| | $^{156}Gd$ | 0.0070440 |
| | $^{157}Gd$ | 0.0054304 |
| | $^{158}Gd$ | 0.0086676 |
| | $^{160}Gd$ | 0.0077296 |
| | $^{16}O$ | 0.0053620 |
| Water | SCALE Standard Composition Library Density = 1.0 g/cm$^3$ @ 293.15 K | |
| Stainless Steel | SCALE Standard Composition Library Density = 7.94 g/cm$^3$ @ 293.15 K | |
| Fresh $UO_2$ | Fraction of Theoretical Density = 0.965 Enrichment = 4.95 w/o $^{235}U$ @ 293.15 K | |
| Regular Concrete | SCALE Standard Composition Library Density = 2.3 g/cm$^3$ @ 293.15 K | |

# 2.0 Design Input

This section provides a brief description of the Prairie Island Units 1 & 2 spent fuel storage racks with the objective of establishing a basis for the analytical models employed in the criticality analyses described in Section 3.0.

## 2.1 Design Input from NMC

In a transmittal documented in Appendix A (Supporting Documentation), NMC provided Westinghouse a comprehensive package of design data related to the Prairie Island Units 1 & 2 spent fuel pools. This design input package includes the necessary data, drawings, or references required to develop the KENO models discussed herein. Specifically, it includes drawing NF-90044 which was employed to develop the KENO model for spent fuel pool #2. The nominal storage cell dimensions were obtained from drawing NF-90046 (also included in the transmittal). Note that drawing NF-90044 contains a typographical error. Module 90047-11 is labeled as a 7x8 module in drawing NF-90044. It is actually a 7x7 module.

## 2.2 Spent Fuel Pool Storage Configuration Description

There are two spent fuel pools which provide storage for Prairie Island Units 1 & 2. Spent fuel pool #1 is the small pool, and spent fuel pool #2 is the large pool. Spent fuel pool #1 contains 9 spent fuel storage modules; there are six 7x7 modules and three 7x8 modules. Spent fuel pool #2 contains 21 spent fuel storage modules; there are seven 7x7 modules, ten 7x8 modules, and four 8x7 modules. The modules are separated by a minimum water gap of 1 inch. Spent fuel pool #2 has a liner inside dimension equal to 227 inches in the north to south direction and 521 inches in the west to east direction. The modules in spent fuel pool #2 are located 2 inches from the southwest corner. Figure 2-2 displays the arrangement of the spent fuel pool storage modules and was produced by scanning drawing NF-90044. Table 2-2 summarizes the overall geometry data for the Prairie Island Units 1 & 2 spent fuel pool #2.

## 2.3 Individual Storage Cell Description

The nominal storage cell is centered on a pitch equal to 9.5 inches. Each storage cell consists of an inner stainless steel canister and outer stainless steel sheathing. The original Boraflex material (not modeled in this analysis) was located in the cavity between the inner canister and outer sheathing. The inner stainless steel canister has a nominal inside dimension equal to 8.27 inches and is 0.09 inches thick. The outer stainless steel sheathing has an inside dimension equal to 8.70 inches and is 0.024 inches thick.

The nominal storage rack dimensions are summarized in Table 2-1. The nominal rack dimensions are reported with manufacturing tolerances, where available. Figure 2-1 displays the Prairie Island Units 1 & 2 storage cell geometry. Figure 2-3 displays the dimensions of the individual storage cell and was produced by scanning drawing NF-90046.

**Table 2-1. Prairie Island Units 1 & 2 Storage Cell Dimensions (All dimensions given in inches)**

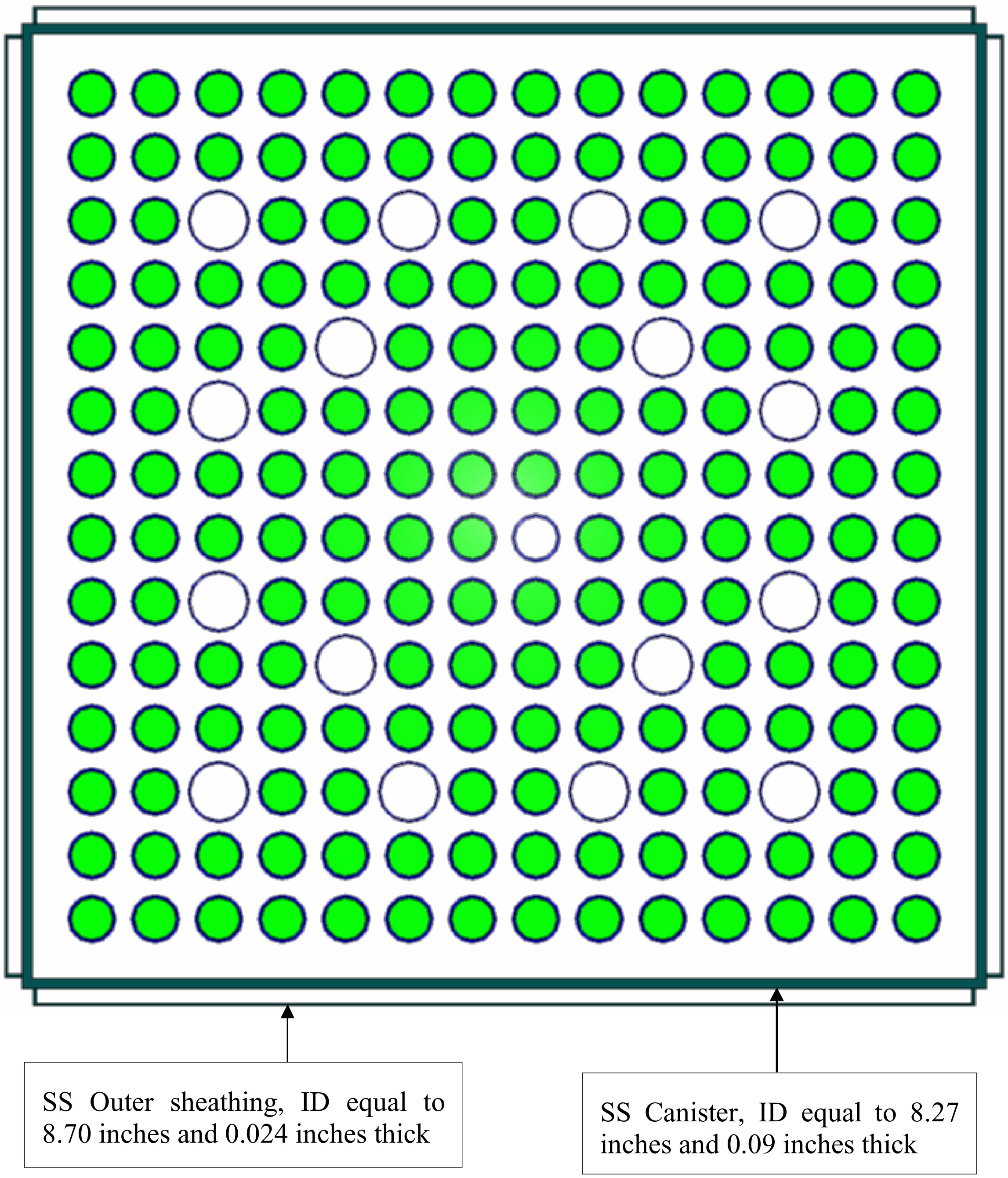
| Parameter | Value |
|---|---|
| Nominal Cell Pitch | 9.50 ± 0.06 |
| Box Wall Thickness | 0.09 ± 0.01 |
| Box ID | 8.27 ± 0.10 |
| Boraflex Cavity Width | 8.20 |
| Boraflex Cavity Thickness | 0.125 |
| Sheathing Thickness | 0.024 |

**Table 2-2. Prairie Island Units 1 & 2 Spent Fuel Pool #2 Dimensions (All dimensions given in inches)**

| **Parameter** | **Value** |
|---|---|
| Pool Length | 227 |
| Pool Width | 521 |
| Intra Module Gap[3] | 1.0 |
| Wall / Module Gap in SW Corner | 2.0 |
| Wall Thickness | 24 |

[3] The intra module water gap is conservatively modeled as 1 inch.

**Figure 2-1. Prairie Island Units 1 & 2 Storage Cell**



# Figure 2-2. Prairie Island Units 1 & 2 Spent Fuel Pool

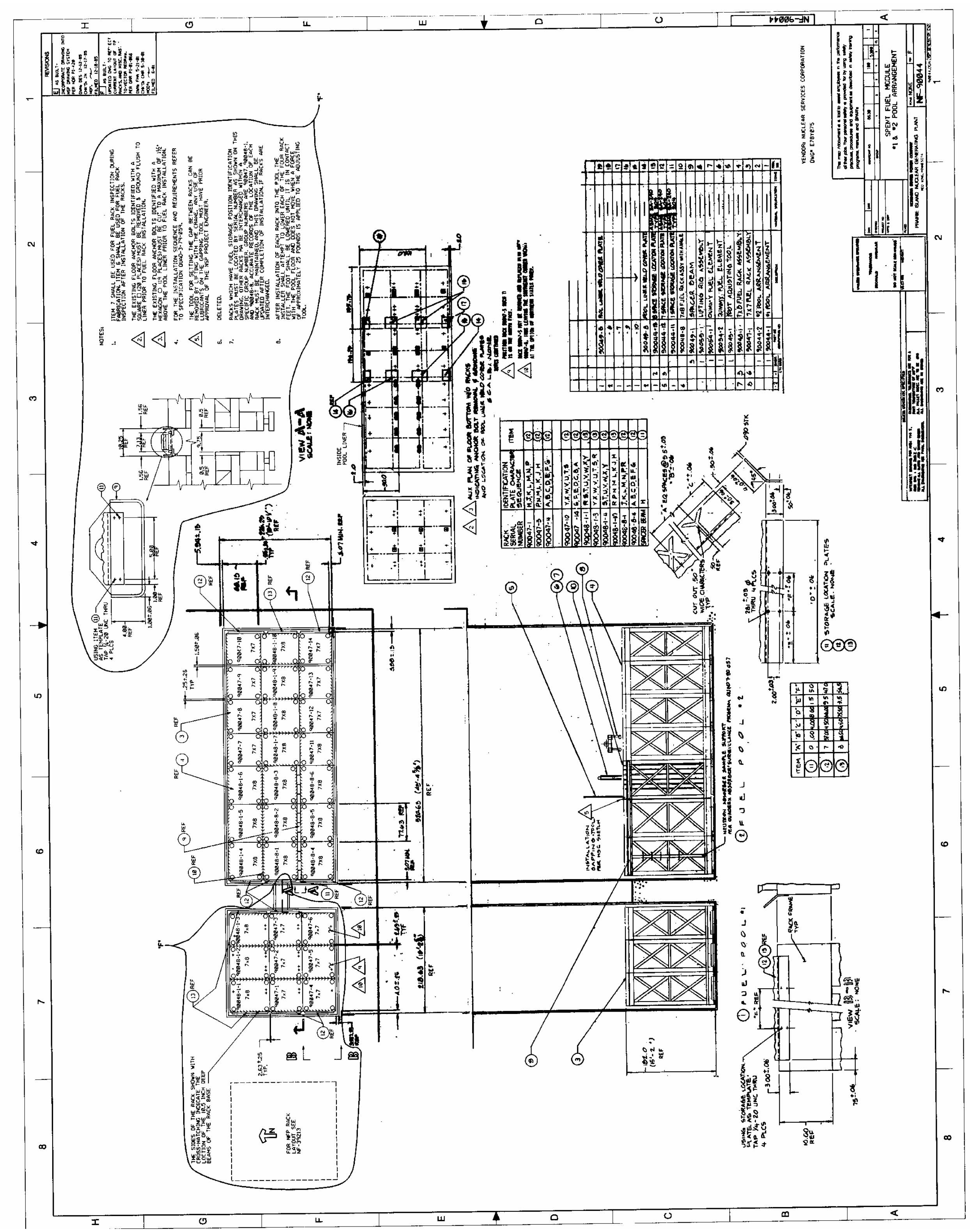
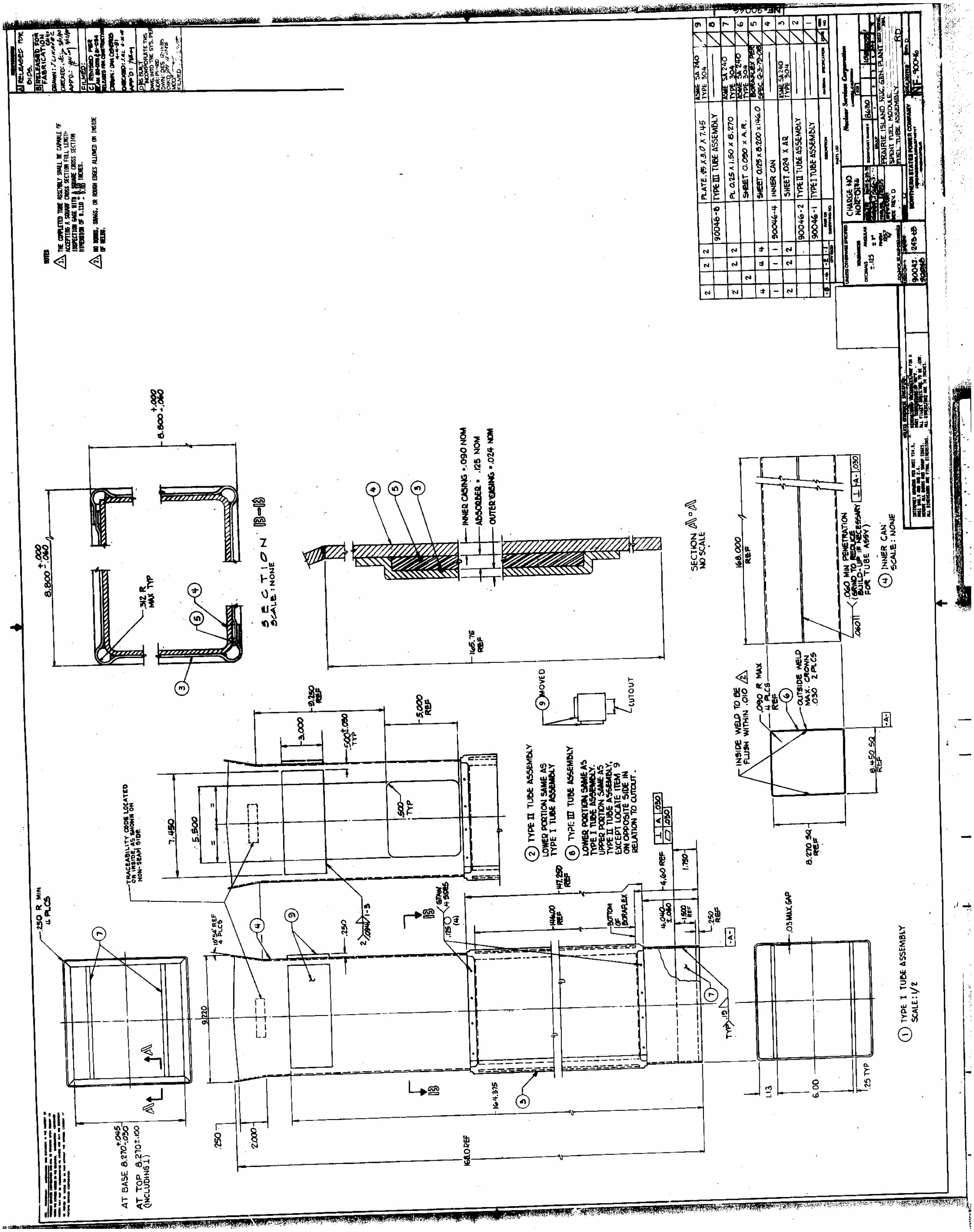


**Figure 2-3. Prairie Island Units 1 & 2 Assembly Storage Cell**



# 3.0 Analysis

## 3.1 KENO Models

The Prairie Island Units 1 & 2 spent fuel storage racks employ two different fuel assembly storage configurations; namely the ”All-Cell” and “3x3” fuel assembly storage configurations. The “3x3” fuel assembly storage configuration is analyzed with and without credit for $Gd_2O_3$ burnable absorbers. The purpose of this section is to describe the models employed in KENO to represent these assembly storage configurations and spent fuel pool #2.

### 3.1.1 KENO Model for the “All-Cell” Fuel Assembly Storage Configuration

An “All-Cell” fuel assembly storage configuration is modeled in KENO as an infinitely repeating storage cell that contains either a fresh or depleted fuel assembly. An inner stainless steel canister controls the fuel assembly position.

Each cell location is modeled in KENO as a square cell with a pitch equal to 9.50 inches. The inner stainless steel canister is modeled with an inside dimension equal to 8.27 inches and is 0.09 inches thick. The outer stainless steel sheathing is modeled with an inside dimension equal to 8.70 inches and is 0.024 inches thick. The cavity between the canister and outer sheathing is modeled with water. All of these dimensions employed to model the Prairie Island Units 1 & 2 storage cell are consistent with the values given in Table 2-1.

The fuel assembly, inner stainless steel canister, and outer stainless steel sheathing are modeled in KENO as 144 inches tall. Reflective boundary conditions are applied to the X and Y surfaces of the assembly, thus simulating an infinitely repeating array. A two-foot water reflector is modeled above and below the storage cell geometry. The pool water is simulated to be full density (1 $g/cm^3$) at room temperature (68 °F). The top and bottom surfaces of the water reflector have reflected boundary conditions.

The fuel assembly modeled in KENO represents the Westinghouse 14x14 Standard design. Note that the fuel pellets in a fuel rod are modeled as a fully enriched right solid cylinder that is 144 inches tall. This assumption conservatively bounds fuel rod designs that incorporate annular and/or lower enrichment fuel pellets such as those employed for axial blankets. A top down image of a KENO produced plot of a single “All-Cell” fuel assembly storage configuration is shown in Figure 3-4.

### 3.1.2 KENO Model for the “3x3” Fuel Assembly Storage Configuration

The “3x3” fuel assembly storage configuration is modeled in KENO as a repeating 3x3 array with a fresh fuel assembly occupying the center location of the array and the remaining locations are occupied by discharged fuel assemblies. This storage

configuration is analyzed with and without credit for $Gd_2O_3$ burnable absorbers. An inner stainless steel canister controls each fuel assembly position within the array.

Each of the nine storage cell locations is modeled in KENO as a square cell with a pitch equal to 9.50 inches. The inner stainless steel canister is modeled with an inside dimension equal to 8.27 inches and is 0.09 inches thick. The outer stainless steel sheathing is modeled with an inside dimension equal to 8.70 inches and is 0.024 inches thick. The cavity between the canister and outer sheathing is modeled with water. All of these dimensions employed to model the Prairie Island Units 1 & 2 storage cell are consistent with the values given in Table 2-1.

The fuel assembly, inner stainless steel canister, and outer stainless steel sheathing are modeled in KENO as 144 inches tall. Reflective boundary conditions are applied to the X and Y surfaces of the 3x3 array, thus simulating an infinitely repeating “3x3” fuel assembly storage configuration. A two-foot water reflector is modeled above and below the storage cell geometry. The pool water is simulated to be full density (1 g/cm$^3$) at room temperature (68 °F). The top and bottom surfaces of the water reflector have reflected boundary conditions.

The center fuel assembly that is modeled in KENO represents the Westinghouse 14x14 OFA design. The enrichment of all fuel pellets is equal to 4.95 w/o $^{235}U$ (with no $Gd_2O_3$ credit) and the pellet density is equal to 96.5% of theoretical density. The remaining fuel assemblies that are modeled by KENO represent the Westinghouse 14x14 Standard design. Note that the fuel pellets in a fuel rod are modeled as fully enriched right solid cylinders that are 144 inches tall. This assumption conservatively bounds fuel rod designs which incorporate annular and/or lower enrichment fuel pellets such as those employed for axial blankets. A top down image of a KENO produced plot of a single “3x3” fuel assembly storage configuration is shown in Figure 3-5.

Storage of fresh fuel assemblies with $Gd_2O_3$ burnable absorbers in the center location of the 3x3 array allow for storage of more reactive fuel assemblies on the periphery than is allowed by the configuration described above. $Gd_2O_3$ credit accounts for the reactivity decrease associated with the addition of a neutron poison material. The following assumptions are used to represent the $Gd_2O_3$ pellets in the KENO model of the 3x3 storage region.

- A 6 inch burnable absorber cutback (top and bottom) is used. This produces a 132 inch shimmed length that is centered about the active fuel height.

- The $Gd_2O_3$ amount is limited to four fuel pins at a concentration of 4.0 w/o $Gd_2O_3$ in $Gd_2O_3$-$UO_2$. The pin placement is shown in Figure 3-2.

- The $^{235}U$ enrichment is reduced to 4.0 w/o in the shimmed portion of the fuel pin for fuel temperature considerations.

- The $^{235}U$ enrichment in the blanket region of the shimmed fuel pins is also reduced to 4.0 w/o.

- The density of the $UO_2$ and $Gd_2O_3$ mixture is found with the following empirical expression (Reference 18),

$$\rho_{U+Gd} = \rho_{U_{th}} \cdot (1 - 0.00301 \cdot \mathrm{X})$$

where,

$$\rho_{U+Gd} \equiv \textit{density of } UO_2 \textit{ and } Gd_2O_3 \textit{ mixture}$$
$$\rho_{U_{th}} \equiv \textit{theoretical } UO_2 \textit{ density}$$
$$\mathrm{X} \equiv Gd_2O_3 \textit{ concentration in weight percentage}$$

The calculation performed for this analysis is as follows,

$$\rho_{U_{th}} = 10.5764\ g \cdot cm^{-3}\ (96.5\%\ T.D.) \quad and \quad \mathrm{X} = 4.0\ w/o,$$
$$\therefore \quad \rho_{U+Gd} = 10.5764 \cdot (1 - 0.00301 \cdot 4.0) = 10.4491\ g \cdot cm^{-3}$$

This value is utilized in the "3x3" storage configuration KENO models with $Gd_2O_3$ credit.

### 3.1.3 KENO Model for Entire Spent Fuel Pool

There is a relatively large amount of leakage in the Prairie Island spent fuel pool #1 (the small pool), therefore only spent fuel pool #2 (the large pool) need be modeled for conservatism. Spent fuel pool #2 is modeled in KENO as a rectangular water cell that is 521 inches in the west to east direction and 227 inches in the north to south direction. The floor and sides of the spent fuel pool are modeled by surrounding the rectangular water cell with two feet of concrete on the bottom and sides.

Twenty one (21) fuel storage modules are inside the spent fuel pool #2 rectangular water cell. The fuel storage modules vary in size from a 7x7 to a 7x8/8x7 array of storage cells. All of the individual assembly storage cells were modeled exactly the same and as described in Sections 3.1.1 through 3.1.2. The minimum intra module water gap of 1.0 inch was modeled conservatively. The fuel storage rack modules are placed within 2.0 inches from the southwest corner of the spent fuel pool liner. Note that a 2 inch gap of water between the modules and pool wall is maintained on the south and west faces and the intra module water gap shown by section A-A in Figure 2-2 is modeled as 9.75 inches wide. These pool dimensions are shown in Table 2-2. The pool water was modeled at room temperature conditions, 68 °F, and as full density (1.0 g/cc).

The storage modules are modeled with both the "All-Cell" and "3x3" storage configurations. Figures 3-6 and 3-7 show KENO produced plots of the spent fuel pool loaded with these storage configurations.These arrangements conform to the restrictions outlined in sections 3.1.1 and 3.1.2. No $Gd_2O_3$ burnable absorber credit is modeled in this portion of the analysis.

### 3.2 Design Basis Fuel Assembly

Prairie Island Units 1 & 2 have been in operation for many years and during that time interval a variety of reload batches containing different fuel assembly designs have been cycled through the reactors. Thus, the criticality safety analysis of their spent fuel pool must take into account possible differences in the reactivity characteristics of the different assembly types. For purposes of this analysis, the different types of fuel assemblies were surveyed so as to define a reference design fuel assembly that would assure conservative results for the analysis.

Table 3-1 provides the relevant dimensions employed to model the Westinghouse 14x14 Standard and Westinghouse 14x14 OFA fuel assemblies in the spent fuel pool environment. Figure 3-1 displays the Westinghouse 14x14 fuel assembly with both the OFA and STD parameters. Based on the results of scoping calculations for the $^{235}U$ loading and storage configuration considered here, the most reactive fresh fuel assembly design is the Westinghouse 14x14 OFA fuel assembly for the center location of the "3x3" fuel assembly storage configuration. The Westinghouse Standard fuel assembly design was modeled as the design basis fuel assembly to conservatively represent discharged fuel assemblies residing in the "All-Cell" and peripheral locations of the "3x3" fuel assembly storage configurations. Other fuel assembly designs are found to be less reactive in these fuel assembly storage configurations than the design basis fuel assemblies.

The unshimmed design basis fuel assemblies are modeled with the maximum enrichment over the active fuel length. The fresh fuel pellets in a fuel rod are modeled as a solid right cylinder with a $UO_2$ density equal to 10.576 g/cc (96.5% of theoretical density). No credit is taken for the nominal 1.1 to 1.2 void fraction percentages that are associated with dishing or chamfering. In addition, no credit is taken for any natural or reduced enrichment pellets. These assumptions result in equivalent or conservative calculations of reactivity for all fuel assemblies used at Prairie Island Units 1 & 2, including those with annular pellets or lower enrichment pellets at the ends of the fuel rods. No credit is taken for any spacer grids or sleeves.

The shimmed fuel assemblies are of the Westinghouse OFA design and incorporate the design features outlined above in section 3.1.2.

### 3.3 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology employed in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the burned fuel assemblies. This effect is important in the analysis of the spent fuel pool characteristics since the majority of spent fuel assemblies stored in the pool have a discharge burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative. Therefore, it is necessary to represent the burnt fuel assembly with a representative axial burnup profile.

For any given spent fuel assembly, the fuel burnup is a continuous function of axial position. However, from a computational point of view, this function can be discretized in such a manner that the axial "end-effect" is adequately captured. It is often common practice to divide the fuel assembly into several axial zones with each zone assumed to be uniform in burnup. Moreover, it is required that the size of the top and bottom axial zones be small (typically less than 8 inches) so as to capture the steep burnup gradient with axial position while that of the central zone may be larger. In spent fuel pool calculations, an eight-zone axial model is found to be adequate (Reference 19) to represent the spent fuel assembly. Such an eight-zone model would have seven zones with fine mesh spacing (four at the top of the fuel assembly and three at the bottom) and the remaining zone represents the center portion of the fuel assembly. Figure 3-3 provides a pictorial view of the axial zones employed in the eight-zone axial model.

The individual power fractions of each zone are so modeled that they give the same volume averaged burnup when compared to a uniform burnup model. This model is validated due to the fact that the relative contribution of the bottom zones of the fuel assembly to the $k_{eff}$ value is negligible.

Input to this analysis is based on the limiting axial burnup profile data provided in the DOE Topical Report, as documented in Reference 14. The burnup profile in the DOE Topical Report is based on a database of 3169 axial-burnup profiles for PWR fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data. The axial burnup profile in the DOE report is based on the most limiting axial burnup shape found in the database. The eight-zone model is constructed based on this limiting axial burnup profile.

DIT was used to generate the isotopic concentrations for each segment of the axial profile. Table 3-2 lists the fuel and moderator temperatures employed in the spectral calculations for each node of the eight-zone axial burnup model. These values are based on mid-cycle temperature profiles for Prairie Island Units 1 and 2. The fuel temperatures for each axial zone are calculated based on a representative fuel temperature correlation while the moderator temperatures are based on a linear relationship with axial position. These node dependent moderator and fuel temperature data and power profile data were employed in DIT to deplete the fuel to the desired burnup for each initial enrichment and

axial zone. The values of assembly average burnups versus feed enrichment for which burned fuel assemblies were simulated are tabulated in Table 3-3.

A constant soluble boron concentration of 800 ppm was employed in all the burnup steps. This value is representative of a cycle average soluble boron concentration in the core. For the purpose of extracting the number densities, the DIT computer code was executed in two modes. First, a normal depletion was continued in steps of 1000 MWD/MTU (with respect to the assembly average case) until the desired burnup was reached. Then a restart is performed at cold, spent fuel pool conditions and the fuel assembly is allowed to decay for 100 hours. At this point of time, the reactivity of the burned fuel assembly is at its highest. The $k_\infty$ and the isotopic number densities are then extracted for the KENO model development at these assembly conditions.

The DIT computed isotopic concentrations were transferred into the KENO models of the storage cells using a limited set of isotopes. That is, the $^{235}U$, $^{238}U$, $^{236}U$, $^{239}Pu$, $^{240}Pu$, $^{241}Pu$, $^{16}O$, and equilibrium $^{149}Sm$ at shutdown are represented explicitly in the KENO models. All other fission product isotopic number densities are represented by an equivalent $^{10}B$ concentration; the magnitude of this concentration is determined by matching the DIT $k_{eff}$ value with the KENO $k_{eff}$ value to a one sigma tolerance level.

Reference 19 contains a listing of the isotopic number densities employed in the KENO calculations. The format of the listing is compatible with the KENO input description and can directly be used as part of KENO input for material specification. The isotopic number densities are listed for the combination of initial enrichment and burnup listed in Table 3-3. The listing is for the Westinghouse 14x14 Standard fuel assembly design.

Reference 19 also contains a listing of the $^{10}B$ number densities determined by matching the DIT $k_{eff}$ and KENO calculated $k_{eff}$ values. The $^{10}B$ number density, the DIT calculated $k_{eff}$ and the KENO calculated $k_{eff}$, for the eight-zone axial model (and the average fuel assembly model) are listed in each table. The first four tables contain these values for 3.0 w/o, the next four tables contain the data for 4.0 w/o, and the final four tables contain data for 5.0 w/o.

### 3.4 Tolerance / Uncertainty Calculations

Previous sections described the storage racks and fuel assembly storage configurations within the spent fuel pool and the KENO models employed to represent repeating arrays of these fuel assembly storage configurations. In addition, the method of modeling the axial profiles of fuel assembly burnup, moderator temperature, and fuel temperature were discussed.

Using the above input, analytic models were developed to perform the quantitative evaluations necessary to demonstrate that the effective multiplication factor for the spent fuel pool is less than 0.999 with zero soluble boron present in the pool water. Applicable biases to be factored into this evaluation are: (1) the methodology bias deduced from the

validation analyses of pertinent critical experiments, and (2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 50 °F to 150 °F (from Reference 20).

A second allowance is based on a 95/95 confidence level assessment of tolerances and uncertainties; included in the summation of variances are the following.

(a) the 95/95 confidence level methods variance,

(b) the 95/95 confidence level calculational uncertainty,

(c) fuel rod manufacturing tolerances,

(d) storage rack fabrication tolerances,

(e) tolerances due to positioning the fuel assembly in the storage cell.

(f) burnup uncertainty

(g) burnable absorber concentration (if applicable)

Items a) and b) are based on the calculational methods validation analyses. For Item c), the fuel rod manufacturing tolerance for the reference design fuel assembly is assumed to consist of two components; an increase in fuel enrichment equal to 0.05 w/o $^{235}U$ and an increase in pellet density from 96.5 to 98.5% of theoretical density; the individual contributions of each change are combined by taking the square root of the sum of the squares of each component. There is no allowance for dishing and chamfer and therefore the pellet density conservatively represents the stack density of the $UO_2$ pellets in the fuel rod.

For item d), the following uncertainty components were evaluated. The inner stainless steel canister ID was increased from 8.27 inches to 8.37 inches and the thickness of the canister was decreased from 0.09 inches to 0.08 inches. The storage cell pitch for the “All-Cell” and “3x3” fuel assembly storage configurations was decreased from 9.50 inches to 9.44 inches.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells (item e), all nominal calculations are carried out with fuel assemblies conservatively centered in the storage cells. One case was run to investigate the effect of off-center position of the fuel assemblies for each of the fuel assembly storage configurations. These cases positioned the assemblies as close as possible in four adjacent storage cells. Eccentric positioning has a slightly positive reactivity effect for all of the fuel assembly storage configurations.

For item f), a 5% burnup uncertainty is included. The 5% burnup uncertainty is applied to the fuel assembly storage configurations that contain depleted fuel assemblies.

For item g), the nominal gadolinia concentration is equal to 4.0 wt %. The tolerance analyzed for the gadolinia concentration is equal to -0.2 wt %.

Table 3-4, Table 3-5, and Table 3-6 provide a summary of the KENO results used in the calculation of biases and uncertainties for the "All-Cell", "Unshimmed 3x3", and "Shimmed 3x3" fuel assembly storage configurations, respectively. The total biases and uncertainties for these fuel assembly storage configurations are 0.02678, 0.02403, and 0.02816 $\Delta k_{eff}$ units respectively.

## 3.5 No Soluble Boron 95/95 $k_{eff}$ Calculational Results

The purpose of the following five subsections is to present the KENO calculated multiplication factors for the "All-Cell" and "3x3" fuel assembly storage configurations along with the result for the entire spent fuel pool at the zero soluble boron condition.

Due to the burnup requirements for storage in these configurations, $^{241}$Pu decay and $^{241}$Am production burnup credit is included. The concentrations for $^{241}$Pu are decayed using the equation below and a half life, $t_{1/2}$, value of 14.4 years.

$$n = n_0 \cdot e^{-\frac{(\ln 2)\cdot t}{t_{1/2}}}$$

The production rate for $^{241}$Am is equal to the rate of $^{241}$Pu decay. The decay time, t, extends 20 years in intervals of 5 years.

### 3.5.1 "All-Cell" Fuel Assembly Storage Configuration

As described in Section 3.1.1, the "All-Cell" fuel assembly storage configuration consists of an infinitely repeating storage cell that contains either fresh or depleted fuel assemblies. The fuel assembly modeled in this analysis is the Westinghouse Standard fuel assembly design.

$k_{eff}$ was evaluated for an infinite array of "All-Cell" storage locations over a range of initial enrichment values up to 5.0 w/o $^{235}$U and assembly average burnups up to 45.0 GWD/T. These calculations were performed at 68 $^{o}$F, with maximum water density equal to 1.0 g/cc, to maximize the array reactivity. KENO calculations were performed for this fuel assembly storage configuration with an axially distributed burnup profile. The relative axial burnup profile employed for these calculations is discussed in Section 3.3. These resulting KENO calculated $k_{eff}$ data are then employed to determine the burnup versus initial enrichment limits for a target $k_{eff}$ value at zero soluble boron. The target value of $k_{eff}$ is selected to be less than 0.999 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses. From Table 3-4, the sum of the biases and uncertainties is equal to 0.02678. Therefore, the target $k_{eff}$ value for the "All-Cell" fuel assembly storage configuration is equal to 0.97222 (0.999-0.02678).

Table 3-7 lists the KENO calculated $k_{eff}$ values for the "All-Cell" fuel assembly storage configuration versus initial enrichment and fuel assembly average burnup for an axially distributed burnup profile. The first entry in each of these tables lists the initial enrichment for no assembly burnup. Based upon the target $k_{eff}$ value, the interpolated enrichment for no assembly burnup is equal to 1.80 w/o $^{235}$U. The derived burnup limits, for enrichments greater than 1.80 w/o $^{235}$U, are based upon the KENO calculated $k_{eff}$ values for 3.0, 4.0, and 5.0 w/o $^{235}$U. For each of these enrichments, KENO calculations were performed at three assembly average burnup values for an axially distributed burnup profile. A second degree fit of the burnup versus $k_{eff}$ data was then employed to determine the burnup required to meet the target $k_{eff}$ value of 0.97222. The resulting assembly burnup versus initial enrichment storage limits are provided in Table 3-10. The first entry in these tables lists the initial enrichment, 1.80 w/o $^{235}$U, for fuel assemblies at zero burnup. The data in this table is plotted in Figure 4-5. The required assembly burnups as a function of initial enrichment were fitted to second degree "least-squares" polynomials. These polynomials are given in Table 4-1 and will be used to determine the burnup as a function of initial enrichment.

### 3.5.2 "3x3" Fuel Assembly Storage Configuration

As described in Section 3.1.2, the "3x3" fuel assembly storage configuration consists of a repeating 3x3 array with a fresh fuel assembly occupying the center location of the array and the remaining locations are occupied by discharged assemblies. The center assembly is the Westinghouse OFA design and the peripheral assemblies are the Westinghouse Standard design. The unshimmed case contains no $Gd_2O_3$ burnable absorbers, and the shimmed case contains four $Gd_2O_3$ burnable absorber pins at a concentration of 4.0 w/o.

$k_{eff}$ was evaluated for an infinite array of "Unshimmed 3x3" storage locations over a range of initial enrichment values up to 5.0 w/o $^{235}$U and assembly average burnups up to 55.0 GWD/T. These calculations were performed at 68 $^{o}$F, with maximum water density equal to 1.0 g/cc, to maximize the array reactivity. KENO calculations were performed for this fuel assembly storage configuration with an axially distributed burnup profile. The relative axial burnup profile employed for these calculations is discussed in Section 3.3. These resulting KENO calculated $k_{eff}$ data are then employed to determine the burnup versus initial enrichment limits for a target $k_{eff}$ value at zero soluble boron. The target value of $k_{eff}$ is selected to be less than 0.999 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses. From Table 3-5, the sum of the biases and uncertainties is equal to 0.02403. Therefore, the target $k_{eff}$ value for the "Unshimmed 3x3" fuel assembly storage configuration is equal to 0.97497 (0.999-0.02403).

Table 3-8 lists the KENO calculated $k_{eff}$ values for the "Unshimmed 3x3" fuel assembly storage configuration versus initial enrichment and fuel assembly average burnup for an axially distributed burnup profile. The first entry in these tables lists the initial enrichment for no assembly burnup. Based upon the target $k_{eff}$ value, the interpolated enrichment for no assembly burnup is equal to 1.30 w/o $^{235}$U. The derived burnup limits,

for enrichments greater than 1.30 w/o $^{235}U$, are based upon the KENO calculated $k_{eff}$ values for 3.0, 4.0, and 5.0 w/o $^{235}U$. For each of these enrichments, KENO calculations were performed at three assembly average burnup values with an axially distributed burnup profile. A second degree fit of the burnup versus $k_{eff}$ data was then employed to determine the burnup required to meet the target $k_{eff}$ value equal to 0.97497. The resulting assembly burnup versus initial enrichment storage limits are provided in Table 3-11. The first entry in these tables lists the initial enrichment, 1.30 w/o $^{235}U$, for fuel assemblies at zero burnup. The data in this table is plotted in Figure 4-6. The required assembly burnups as a function of initial enrichment were fitted to second degree "least-squares" polynomials. These polynomials are given in Table 4-2 and will be used to determine the burnup as a function of initial enrichment.

$k_{eff}$ was also evaluated for an infinite array of "Shimmed 3x3" storage locations over a range of initial enrichment values up to 5.0 w/o $^{235}U$ and assembly average burnups up to 45.0 GWD/T. These calculations were performed at 68 $^{o}$F, with maximum water density equal to 1.0 g/cc, to maximize the array reactivity. KENO calculations were performed for this fuel assembly storage configuration with an axially distributed burnup profile. The relative axial burnup profile employed for these calculations is discussed in Section 3.3. These resulting KENO calculated $k_{eff}$ data are then employed to determine the burnup versus initial enrichment limits for a target $k_{eff}$ value at zero soluble boron. The target value of $k_{eff}$ is selected to be less than 0.999 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses. From Table 3-6, the sum of the biases and uncertainties is equal to 0.02816. Therefore, the target $k_{eff}$ value for the "shimmed 3x3" fuel assembly storage configuration is equal to 0.97084 (0.999-0.02816).

Table 3-9 lists the KENO calculated $k_{eff}$ values for the "Shimmed 3x3" fuel assembly storage configuration versus initial enrichment and fuel assembly average burnup for both a uniform and axially distributed burnup profile. The first entry in these tables lists the initial enrichment for no assembly burnup. Based upon the target $k_{eff}$ value, the interpolated enrichment for no assembly burnup is equal to 1.46 w/o $^{235}U$. The derived burnup limits, for enrichments greater than 1.46 w/o $^{235}U$, are based upon the KENO calculated $k_{eff}$ values for 3.0, 4.0, and 5.0 w/o $^{235}U$. For each of these enrichments, KENO calculations were performed at three assembly average burnup values for an axially distributed burnup profile. A second degree fit of the burnup versus $k_{eff}$ data was then employed to determine the burnup required to meet the target $k_{eff}$ value equal to 0.97084. The resulting assembly burnup versus initial enrichment storage limits are provided in Table 3-12. The first entry in these tables lists the initial enrichment, 1.46 w/o $^{235}U$, for fuel assemblies at zero burnup. The data in this table is plotted in Figure 4-7. The required assembly burnups as a function of initial enrichment were fitted to second degree "least-squares" polynomials. These polynomials are given in Table 4-3 and will be used to determine the burnup as a function of initial enrichment.

### 3.5.3 Entire Spent Fuel Pool

KENO models for the entire Prairie Island spent fuel pool #2 were constructed for this analysis and are shown in Figures 3-6 and 3-7. Figure 3-6 displays the KENO model for spent fuel #2, based upon the “All-Cell” storage configuration, with the 2 inch wall gap maintained on the south and west sides of spent fuel pool #2.. Figures 3-7 illustrate the same KENO models based upon the “Unshimmed 3x3” storage configuration. These spent fuel pool KENO models are described in section 3.1.3. The largest KENO calculated multiplication factors for the spent fuel pool models and the respective infinite array models are shown in Table 3-13, and are based upon no soluble boron. The differences in the infinite array and spent fuel pool model’s $k_{eff}$ values are attributed to neutron leakage from the spent fuel #2 model. The biases and uncertainties, from Table 3-4 and Table 3-5, were added to the spent fuel pool multiplication factors and the results are shown in Table 3-13. As can be seen from Table 3-13, the final $k_{95/95}$ values at zero soluble for spent fuel pool #2 are all below the design basis limit equal to 0.999 at zero soluble boron.

The interface between the “All-Cell” and “3x3” storage configurations was directly simulated in a KENO model for spent fuel pool #2. The interface modeled is depicted in Figure 4-4. Note that the KENO calculated multiplication factor for this interface model is 0.96346 +/- 0.00038. This value is less than the value given in Table 3-13 for the “3x3” storage configuration. Therefore, the interface configuration (with biases and uncertainties) also meets the design basis limit equal to 0.999 at zero soluble boron.

## 3.6 Soluble Boron

The NRC Safety Evaluation Report (SER) for WCAP-14416-P is given in reference 3; page 9 of the enclosure to reference 3 defines the soluble boron requirement as follows. The total soluble boron credit requirement is defined as the sum of three quantities.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where,

$SBC_{TOTAL}$ is the total soluble boron credit requirement,(ppm),

$SBC_{95/95}$ is the soluble boron requirement for 95/95 $k_{eff}$ less than or equal to 0.95, (ppm),

$SBC_{RE}$ is the soluble boron required for burnup and reactivity uncertainties, (ppm),

$SBC_{PA}$ is the soluble boron required for $k_{eff}$ less than or equal to 0.95 under accident conditions, (ppm).

Each of these terms will be discussed in the following subsections.

### 3.6.1 Soluble Boron Requirement to Maintain $k_{eff}$ Less Than or Equal to 0.95

Table 3-14 contains the KENO calculated $k_{eff}$ values for the Prairie Island Units 1 & 2 spent fuel pool #2 from 0 to 600 ppm of soluble boron, in increments of 200 ppm. These KENO models assume that the pool is filled with the geometries and storage configurations outlined in section 3.1.3. The reactivity worth, $\Delta k_{eff}$ , of the soluble boron was determined by subtracting the $k_{eff}$ value, for a given soluble boron concentration, from the $k_{eff}$ value for zero soluble boron. The soluble boron concentration and reactivity worth data was then fitted to a second degree polynomial, the limiting of which is shown on the bottom of Table 3-14. This polynomial was then employed to determine the amount of soluble boron required to reduce $k_{eff}$ by 0.05 $\Delta k_{eff}$ units, which is 276 ppm.

### 3.6.2 Soluble Boron Requirement for Burnup and Reactivity Uncertainties

The soluble boron credit, in units of ppm, required for reactivity uncertainties was determined by converting the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel assembly burnup values to a soluble boron concentration, in units of ppm, necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly reactivity, is calculated by employing a depletion reactivity uncertainty equal to 0.010 $\Delta k_{eff}$ units per 30,000 MWD/MTU of assembly burnup (obtained from Reference 3) and multiplying by the maximum amount of assembly burnup credited in a storage region analysis. For this analysis, the maximum amount of assembly burnup credited is 50,281.8 MWD/MTU (for the “Unshimmed 3x3” storage configuration). Therefore, the depletion reactivity uncertainty is 0.016761 $\Delta k_{eff}$.

The uncertainty in absolute fuel assembly burnup values is conservatively calculated as 5% of the maximum fuel assembly burnup credited in a storage region analysis. The maximum fuel assembly burnup credited in the storage configurations considered here, the uncertainty in these burnup values, and the corresponding reactivity values are given in Table 3-15. The reactivity associated with a change in burnup of 2,250 MWD/MTU at 45,000 MWD/MTU for the “All-Cell” storage region was conservatively calculated to be 0.01016 $\Delta k_{eff}$ units.

The total of these two reactivity effects is equal to 0.026921 (0.016761 + 0.01016) $\Delta k_{eff}$ units. The soluble boron concentration (ppm) necessary to compensate for this reactivity was conservatively calculated to be 183 ppm.

### 3.6.3 Soluble Boron Required to Mitigate Accidents

The soluble boron concentration, in units of ppm, required to maintain $k_{eff}$ less than or equal to 0.95 under accident conditions is determined by first surveying all possible events which increase the $k_{eff}$ value of the spent fuel pool. The accident event which produced the largest increase in spent fuel pool $k_{eff}$ value is employed to determine the

required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered include:

Dropped fresh fuel assembly on top of the storage racks,

Misloaded fresh fuel assembly into incorrect storage rack location,

Misloaded fresh fuel assembly between storage racks (in gap between storage racks),

Intramodule water gap reduction due to seismic event,

Spent fuel pool temperature greater than 150 °F.

It is possible to drop a fresh fuel assembly on top of the spent fuel pool storage racks. In this case the physical separation between the fuel assemblies in the spent fuel pool storage racks and the assembly lying on top of the racks is sufficient to neutronically decouple the accident. In other words, dropping the fresh fuel assembly on top of the storage racks will only produce a very small positive reactivity increase. This very small positive reactivity increase will not be as limiting as the reactivity increase associated with fuel mishandling events.

Several fuel mishandling events were simulated with KENO to assess the possible increase in the $k_{eff}$ value of the Prairie Island Units 1 & 2 spent fuel pool #2. The fuel mishandling events all assumed that a fresh Westinghouse OFA fuel assembly enriched to 4.95 w/o $^{235}U$ (and no burnable poisons) was misloaded into the described area of the spent fuel pool. These cases were simulated with the KENO model for spent fuel pool #2. These cases involved placing a fresh fuel assembly either inside a storage location intended for a burned fuel assembly or inside the gap of water between the storage modules in the southwest corner of spent fuel pool #2. The results of these KENO cases are contained in Table 3-16 which indicates that the highest increase in reactivity occurred when a fresh fuel assembly was placed in the gap of water between storage modules and next to another fresh fuel assembly. The reactivity increase associated with this accident was calculated to be 0.05914 $\Delta k_{eff}$ units. The amount of soluble boron necessary to mitigate the consequence of this accident was determined to be 263 ppm by performing a KENO case for the same accident at 300 ppm and linear interpolation of the soluble boron for a reduction of 0.05914 $\Delta k_{eff}$ units.

For the accident due to a seismic event the intramodule water gap is reduced to zero and each storage module makes contact. Based upon the comparison of the $k_{eff}$ values for the entire spent fuel pool and infinite arrays (see Table 3-13) the reactivity associated with this accident is approximately 0.011 $\Delta k_{eff}$ units, and therefore not as limiting as the fuel mishandling events discussed above.

For the change in spent fuel pool water temperature accident, a temperature range of 150 F to 240 F was considered. From page 20 of Reference 20, the maximum reactivity

increase occurred for the "All-Cell" storage configuration and was calculated to be 0.01729 $\Delta k_{eff}$ units. This reactivity increase is far less limiting than the reactivity increase associated with the fuel mishandling events discussed above.

### 3.6.4 Total Soluble Boron Requirement

Soluble boron in the spent fuel pool coolant is used in this criticality safety analysis to offset the reactivity allowances for calculational uncertainties in modeling, storage rack fabrication tolerances, fuel assembly design tolerances, and postulated accidents. The total soluble boron requirement is defined above.

The magnitude of each soluble boron requirement is shown below.

$SBC_{95/95}$ = 276 ppm

$SBC_{RE}$ = 183 ppm

$SBC_{PA}$ = 263 ppm

$SBC_{TOTAL}$ = 722 ppm

Therefore, a total of 722 ppm of soluble boron is required to maintain $k_{eff}$ less than or equal to 0.95 (including all biases and uncertainties) assuming the most limiting single accident. Note that these soluble boron concentrations assumes an atomic fraction for $^{10}B$ equal to 0.199. For a $^{10}B$ isotopic fraction equal to 0.197, the soluble boron concentrations, required to maintain the same concentration of $^{10}B$ atoms, would be calculated as below.

$SBC_{95/95}$ = 279 ppm

$SBC_{RE}$ = 185 ppm

$SBC_{PA}$ = 266 ppm

$SBC_{TOTAL}$ = 730 ppm

Thus a recommended soluble boron level of 730 ppm is sufficient to accommodate all the design requirements.

**Table 3-1. Summary of Fuel Assembly Characteristics (from Reference 20)**

| Characteristics | Westinghouse Standard | Westinghouse OFA |
|---|---|---|
| Cell Pitch (in) | 0.556 | 0.556 |
| Pellet OD (in) | .3659* | .3444 |
| Fuel Rod Clad ID (in) | .3734 | .3514 |
| Fuel Rod Clad OD (in) | .422 | .400 |
| Fuel Rod Clad Material | Zirc-4 | Zirc-4/ Zirlo** |
| Guide Tube ID (in) | .505 | .492 |
| Guide Tube OD (in) | .539 | .526 |
| Instrument Tube ID | 0.374 | 0.352 |
| Instrument Tube OD | 0.422 | 0.399 |
| Enrichment w/o $U_{235}$ | 5.00 | 5.00 |

* Note that 0.3669 inches was conservatively employed to represent this pellet diameter

** Note that the clad material was conservatively modeled as Zr-4.

**Table 3-2. Relative Power, Fuel, and Moderator Temperatures for Eight Zone Model**

| Zone No. | Height (in.) | Relative Power | Fuel Temperature (°F) | Moderator Temperature (°F) |
|---|---|---|---|---|
| 1 | 6.15 | 0.488 | 991.022 | 544.190 |
| 2 | 6.15 | 0.813 | 1101.020 | 545.018 |
| 3 | 6.15 | 1.003 | 1211.018 | 545.360 |
| 4 | 107.1 | 1.092 | 1218.956 | 574.034 |
| 5 | 6.15 | 0.936 | 1138.010 | 603.860 |
| 6 | 3.075 | 0.841 | 1085.522 | 604.526 |
| 7 | 6.15 | 0.624 | 980.528 | 605.741 |
| 8 | 3.075 | 0.297 | 875.516 | 606.488 |

**Table 3-3. Burnup and Initial Enrichment Combinations Used to Determine the Isotopic Number Densities**

| **3 w/o $^{235}$U** | **4 w/o $^{235}$U** | **5 w/o $^{235}$U** |
|---|---|---|
| [MWD/MTU] | [MWD/MTU] | [MWD/MTU] |
| 5,000 | 15,000 | 25,000 |
| 15,000 | 25,000 | 35,000 |
| 25,000 | 35,000 | 45,000 |
| 35,000 | 45,000 | 55,000 |

**Table 3-4. $k_{eff}$ for the Various Physical Tolerance Cases for the "All-Cell" Storage Configuration**

| **Case Description** | **$k_{eff}$** | **$\Delta k_{eff}$** |
|---|---|---|
| 1.80 w/o Nominal Case[4] | $0.97157 \pm 0.00033$ | |
| 4.95 w/o Nominal Case[5] | $1.23265 \pm 0.00038$ | |
| Increase in $^{235}U$ Enrichment | $1.23368 \pm 0.00038$ | 0.00179 |
| Increase in Stack Density | $1.23299 \pm 0.00038$ | 0.00110 |
| Decrease in Cell Pitch | $0.97962 \pm 0.00034$ | 0.00872 |
| Decrease in Rack Thickness | $0.97769 \pm 0.00033$ | 0.00678 |
| Decrease in Rack ID | $0.97364 \pm 0.00034$ | 0.00274 |
| Off-Center Assembly Positioning | $0.97703 \pm 0.00033$ | 0.00612 |
| Burnup Uncertainty | | 0.01016 |
| Methodology Uncertainty[6] | | 0.00646 |
| *Statistical Sum of Uncertainties* | | 0.01779 |
| Methodology Bias[7] | | 0.00259 |
| Pool Temperature Bias[8] | | 0.00640 |
| **Sum of Uncertainties and Biases** | | 0.02678 |

[4] Note the 1.80 w/o nominal KENO case for the All Cell storage contains STD fuel at the fresh enrichment of 1.80 $^{w}/_{o}$ $^{235}U$.
[5] Note the 4.95 w/o nominal KENO case for the All Cell storage contains STD fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$.
[6] See page 11 for definition of methodology uncertainty
[7] Methodology bias or the mean calculational methods bias is evaluated to be 0.00259.
[8] Pool temperature bias obtained from Reference 20.

**Table 3-5. $k_{eff}$ for the Various Physical Tolerance Cases for the Unshimmed "3x3" Storage Configuration**

| **Case Description** | $k_{eff}$ | $\Delta k_{eff}$ |
|---|---|---|
| 1.20 w/o Nominal Case[9] | 0.96471 ± 0.00036 | |
| 4.95 w/o Nominal Case[10] | 1.23181 ± 0.00037 | |
| Increase in $^{235}U$ Enrichment | 1.23350 ± 0.00038 | 0.00244 |
| Increase in Stack Density | 1.23243 ± 0.00040 | 0.00139 |
| Decrease in Cell Pitch | 0.96903 ± 0.00035 | 0.00503 |
| Decrease in Rack Thickness | 0.96752 ± 0.00036 | 0.00353 |
| Increase in Rack ID | 0.96498 ± 0.00037 | 0.00100 |
| Off-Center Assembly Positioning | 0.97372 ± 0.00036 | 0.00973 |
| Burnup Uncertainty | | 0.00658 |
| Methodology Uncertainty[11] | | 0.00646 |
| *Statistical Sum of Uncertainties* | | 0.01504 |
| Methodology Bias[12] | | 0.00259 |
| Pool Temperature Bias[13] | | 0.00640 |
| **Sum of Uncertainties and Biases** | | 0.02403 |

[9] Note the 1.20 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$, which is surrounded by a ring of STD fuel at the fresh enrichment of 1.20 $^{w}/_{o}$ $^{235}U$.

[10] Note the 4.95 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$, which is surrounded by a ring of STD fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$.

[11] See page 11 for definition of methodology uncertainty.

[12] Methodology bias or the mean calculational methods bias is evaluated to be 0.00259.

[13] Pool temperature bias obtained from Reference 20.

**Table 3-6. $k_{eff}$ for the Various Physical Tolerance Cases for the Shimmed "3x3" Storage Configuration**

| **Case Description** | $k_{eff}$ | $\Delta k_{eff}$ |
|---|---|---|
| 1.50 w/o Nominal Case[14] | 0.97619 ± 0.00034 | |
| 4.95 w/o Nominal Case[15] | 1.22695 ± 0.00039 | |
| Increase in $^{235}U$ Enrichment | 1.22923 ± 0.00039 | 0.00306 |
| Increase in Stack Density | 1.22880 ± 0.00039 | 0.00263 |
| Decrease in Cell Pitch | 0.98347 ± 0.00035 | 0.00797 |
| Decrease in Rack Thickness | 0.98016 ± 0.00035 | 0.00466 |
| Increase in Rack ID | 0.97704 ± 0.00035 | 0.00154 |
| Off-Center Assembly Positioning | 0.98483 ± 0.00035 | 0.00933 |
| Decrease in $Gd_2O_3$ Concentration | 0.97588 ± 0.00035 | 0.00038 |
| Burnup Uncertainty | | 0.00752 |
| Methodology Uncertainty[16] | | 0.00646 |
| *Statistical Sum of Uncertainties* | | 0.01916* |
| Methodology Bias[17] | | 0.00259 |
| Pool Temperature Bias[18] | | 0.00640 |
| **Sum of Uncertainties and Biases** | | 0.02816 |

* Conservative, actual value is 0.01701 delta k-effective units.

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[14] Note the 1.50 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$, which is surrounded by a ring of STD fuel at the fresh enrichment of 1.50 $^{w}/_{o}$ $^{235}U$.

[15] Note the 4.95 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$, which is surrounded by a ring of STD fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ $^{235}U$.

[16] See page 11 for definition of methodology uncertainty.

[17] Methodology bias or the mean calculational methods bias is evaluated to be 0.00259.

[18] Pool temperature bias obtained from Reference 20.

**Table 3-7. $k_{eff}$ versus Initial Enrichment and Assembly Burnup for the "All-Cell" Storage Configuration with No Soluble Boron**

| Initial Enrichment (w/o $^{235}$U) | Assembly Burnup (MWD/MTU) | $k_{eff}$ Value Decay Time (years) | | | | |
|---|---|---|---|---|---|---|
| | | 0 | 5 | 10 | 15 | 20 |
| 1.8 | 0 | N/A | N/A | N/A | N/A | N/A |
| 3.0 | 5,000 | 1.05509 | 1.05296 | 1.05257 | 1.05203 | 1.05189 |
| 3.0 | 15,000 | 0.96342 | 0.95647 | 0.95173 | 0.94810 | 0.94511 |
| 3.0 | 25,000 | 0.89586 | 0.88528 | 0.87621 | 0.87085 | 0.86581 |
| 4.0 | 15,000 | 1.04167 | 1.03695 | 1.03345 | 1.03029 | 1.02818 |
| 4.0 | 25,000 | 0.97260 | 0.96512 | 0.95878 | 0.95428 | 0.95044 |
| 4.0 | 35,000 | 0.91680 | 0.90598 | 0.89802 | 0.89109 | 0.88668 |
| 5.0 | 25,000 | 1.03293 | 1.02597 | 1.02081 | 1.01745 | 1.01429 |
| 5.0 | 35,000 | 0.97891 | 0.97055 | 0.96311 | 0.95804 | 0.95357 |
| 5.0 | 45,000 | 0.93132 | 0.92004 | 0.91199 | 0.90535 | 0.90018 |

**Table 3-8. $k_{eff}$ versus Initial Enrichment and Assembly Burnup for the Unshimmed "3x3" Storage Configuration**

| Initial Enrichment (w/o $^{235}U$) | Assembly Burnup (MWD/MTU) | $k_{eff}$ Value | | | | |
|---|---|---|---|---|---|---|
| | | Decay Time (years) | | | | |
| | | 0 | 5 | 10 | 15 | 20 |
| 1.3 | 0 | N/A | N/A | N/A | N/A | N/A |
| 3.0 | 15,000 | 1.02206 | 1.01827 | 1.01439 | 1.01271 | 1.01091 |
| 3.0 | 25,000 | 0.98138 | 0.97613 | 0.97112 | 0.96693 | 0.96371 |
| 3.0 | 35,000 | 0.95284 | 0.94771 | 0.94381 | 0.93758 | 0.93601 |
| 4.0 | 25,000 | 1.02507 | 1.01832 | 1.01376 | 1.00931 | 1.00722 |
| 4.0 | 35,000 | 0.98691 | 0.97992 | 0.97508 | 0.97023 | 0.96734 |
| 4.0 | 45,000 | 0.95919 | 0.95277 | 0.94768 | 0.94336 | 0.94058 |
| 5.0 | 35,000 | 1.02408 | 1.01798 | 1.01123 | 1.00702 | 1.00468 |
| 5.0 | 45,000 | 0.99148 | 0.98192 | 0.97837 | 0.97205 | 0.96932 |
| 5.0 | 55,000 | 0.96393 | 0.95477 | 0.95042 | 0.94501 | 0.94286 |

**Table 3-9. $k_{eff}$ versus Initial Enrichment and Assembly Burnup for the Shimmed "3x3" Storage Configuration**

| Initial Enrichment (w/o $^{235}$U) | Assembly Burnup (MWD/MTU) | $k_{eff}$ Value Decay Time (years) | | | | |
|---|---|---|---|---|---|---|
| | | 0 | 5 | 10 | 15 | 20 |
| 1.46 | 0 | N/A | N/A | N/A | N/A | N/A |
| 3.0 | 5,000 | 1.07394 | 1.07387 | 1.07322 | 1.07240 | 1.07205 |
| 3.0 | 15,000 | 1.00569 | 1.00044 | 0.99736 | 0.99285 | 0.99075 |
| 3.0 | 25,000 | 0.95830 | 0.95047 | 0.94594 | 0.94165 | 0.93888 |
| 4.0 | 15,000 | 1.06381 | 1.05964 | 1.05611 | 1.05356 | 1.05145 |
| 4.0 | 25,000 | 1.00643 | 0.99921 | 0.99372 | 0.98953 | 0.98634 |
| 4.0 | 35,000 | 0.96284 | 0.95564 | 0.94981 | 0.94696 | 0.94371 |
| 5.0 | 25,000 | 1.05250 | 1.04664 | 1.04146 | 1.03788 | 1.03478 |
| 5.0 | 35,000 | 1.00477 | 0.99767 | 0.99045 | 0.98597 | 0.98254 |
| 5.0 | 45,000 | 0.96744 | 0.96138 | 0.95526 | 0.95177 | 0.94908 |

**Table 3-10. Fuel Assembly Burnup versus Initial Enrichment for the "All Cell" Storage Configuration**

| Initial Enrichment (w/o $^{235}$U) | Burnup (MWD/MTU) | | | | |
|---|---|---|---|---|---|
| | Decay Time (years) | | | | |
| | 0 | 5 | 10 | 15 | 20 |
| 1.8 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| 3.0 | 13459.3 | 12715.2 | 12307.9 | 11989.1 | 11758.8 |
| 4.0 | 24542.7 | 23423.8 | 22555.1 | 21998.4 | 21526.0 |
| 5.0 | 35698.3 | 34082.9 | 32754.5 | 31937.0 | 31235.6 |

**Table 3-11. Fuel Assembly Burnup versus Initial Enrichment for the Unshimmed "3x3" Storage Configuration**

| Initial Enrichment (w/o $^{235}U$) | Burnup (MWD/MTU) | | | | |
|---|---|---|---|---|---|
| | Decay Time (years) | | | | |
| | 0 | 5 | 10 | 15 | 20 |
| 1.3 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| 3.0 | 26549.6 | 24899.4 | 23376.5 | 22354.9 | 21420.7 |
| 4.0 | 38501.7 | 36140.4 | 34560.9 | 33044.3 | 32194.1 |
| 5.0 | 50281.8 | 46842.9 | 45644.9 | 43558.8 | 42671.7 |

**Table 3-12. Fuel Assembly Burnup versus Initial Enrichment for the Shimmed "3x3" Storage Configuration**

| Initial Enrichment (w/o $^{235}U$) | Burnup (MWD/MTU) | | | | |
|---|---|---|---|---|---|
| | Decay Time (years) | | | | |
| | 0 | 5 | 10 | 15 | 20 |
| 1.46 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| 3.0 | 22119.5 | 20619.7 | 19840.6 | 18964.7 | 18512.6 |
| 4.0 | 33019.7 | 31255.7 | 29914.7 | 29070.0 | 28328.7 |
| 5.0 | 44022.3 | 42192.6 | 40271.4 | 39100.4 | 38187.8 |

**Table 3-13. Entire Spent Fuel Pool #2 and Infinite Array $k_{eff}$ Results for the Allowable Storage Configurations**

| Configuration Description | Without Biases & Uncertainties | | With Biases & Uncertainties |
|---|---|---|---|
| | Entire Pool $k_{eff}$ | Infinite Array $k_{eff}$ | Entire Pool $k_{eff}$ |
| All-Cell, 1.80 w/o $^{235}U$ and zero burnup | 0.96045 | 0.97157 | 0.98723 |
| Unshimmed 3x3, 1.20 w/o $^{235}U$ and zero burnup | 0.96647 | 0.97477 | 0.99050 |

**Table 3-14. $k_{eff}$ as a Function of Soluble Boron Level**

| | "All-Cell" Storage Configuration | | "3x3" Storage Configuration | |
|---|---|---|---|---|
| **(ppm)** | **$k_{eff}$** | **$\Delta k_{eff}$** | **$k_{eff}$** | **$\Delta k_{eff}$** |
| 0 | 0.92117 | 0.00000 | 0.95695 | 0.00000 |
| 200 | 0.88289 | 0.03828 | 0.91989 | 0.03706 |
| 400 | 0.84953 | 0.07164 | 0.88657 | 0.07038 |
| 600 | 0.82076 | 0.10041 | 0.85623 | 0.10072 |

The most limiting case is the "3x3" storage configuration. The following second degree polynomial describes the soluble boron concentration as a function of $\Delta k_{eff}$:

$$\boxed{\mathbf{ppm = 8894.9\ \Delta k_{eff}^2 + 5059.8\ \Delta k_{eff}}}$$

**Table 3-15. Reactivity Associated with 5 % Burnup Uncertainty for the Storage Configurations**

| Configuration | Maximum BU Considered (MWD/MTU) | 5% BU Uncertainty | $\Delta k_{eff}$ |
|---|---|---|---|
| All-Cell | 45,000 | 2,250 | 0.01016 |
| Unshimmed 3x3 | 50,300 | 2,500 | 0.00658 |
| Shimmed 3x3 | 45,000 | 2,250 | 0.00752 |

**Table 3-16. $k_{eff}$ for Accident Events**

| | All Cell | | 3x3 | |
|---|---|---|---|---|
| **Description of Accident** | **$\Delta k_{eff}$** | **$\Delta$ppm** | **$\Delta k_{eff}$** | **$\Delta$ppm** |
| Dropped fresh fuel assembly on top of the storage racks, | Not Limiting | Not Limiting | Not Limiting | Not Limiting |
| Misloaded fresh fuel assembly into burned storage rack location, | 0.04158 | 179.5 | 0.05336 | 244.2 |
| Misloaded fresh fuel assembly between storage racks, | 0.03207 | 112.3 | 0.05914 | 262.5 |
| Intramodule water gap reduction due to seismic event, | Not Limiting | Not Limiting | Not Limiting | Not Limiting |
| Spent fuel pool temperature greater than 185 °F | Not Limiting | Not Limiting | Not Limiting | Not Limiting |

**Figure 3-1. Westinghouse 14x14 OFA & STD Fuel Assembly**

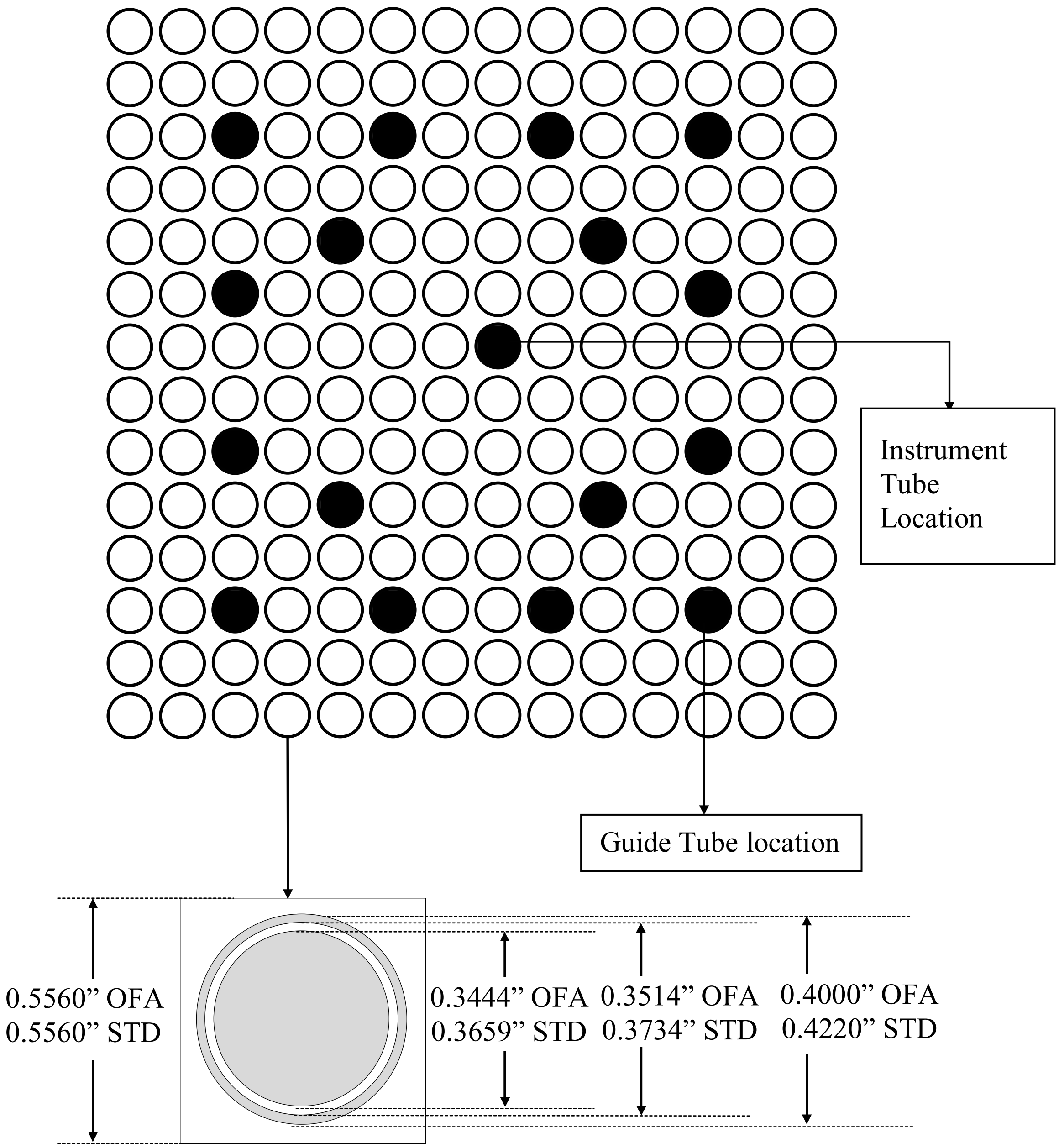


**Figure 3-2. $Gd_2O_3$ Burnable Absorber Pin Pattern**

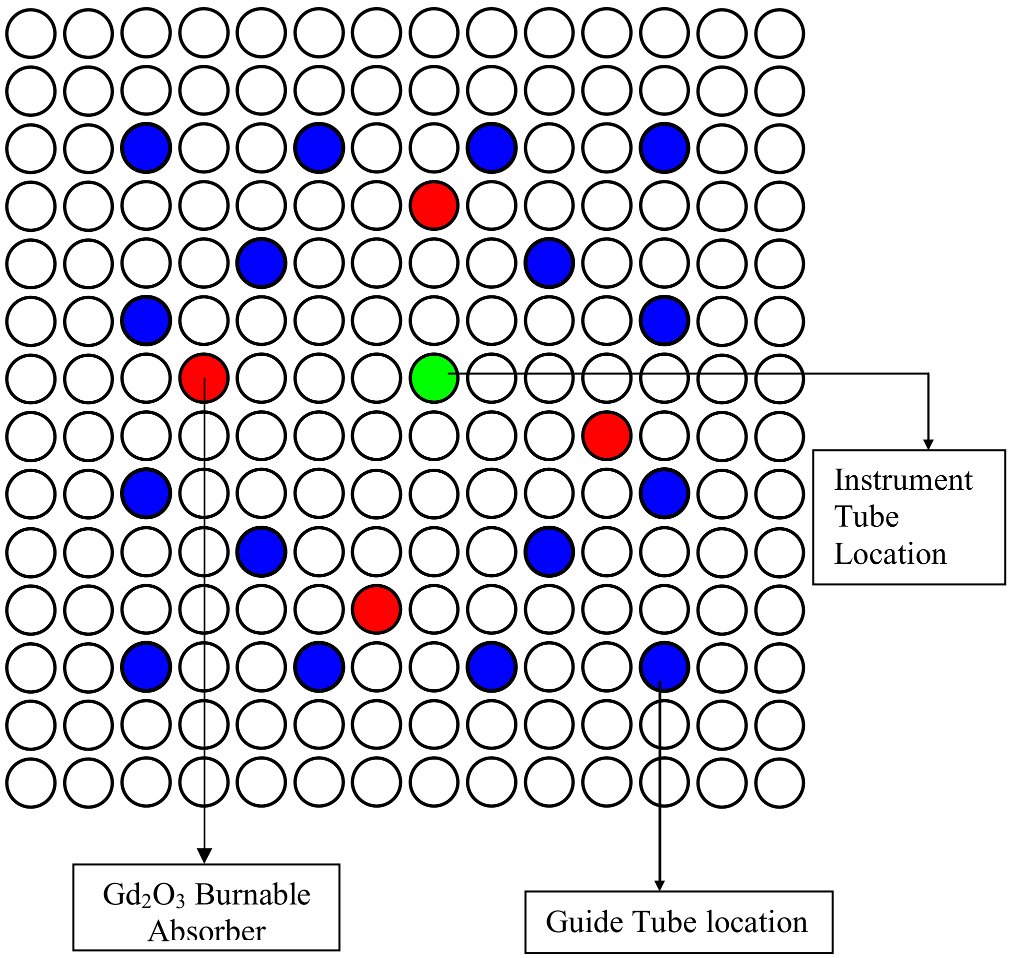


**Figure 3-3. Sketch of Axial Zones Employed in Fuel Assembly**

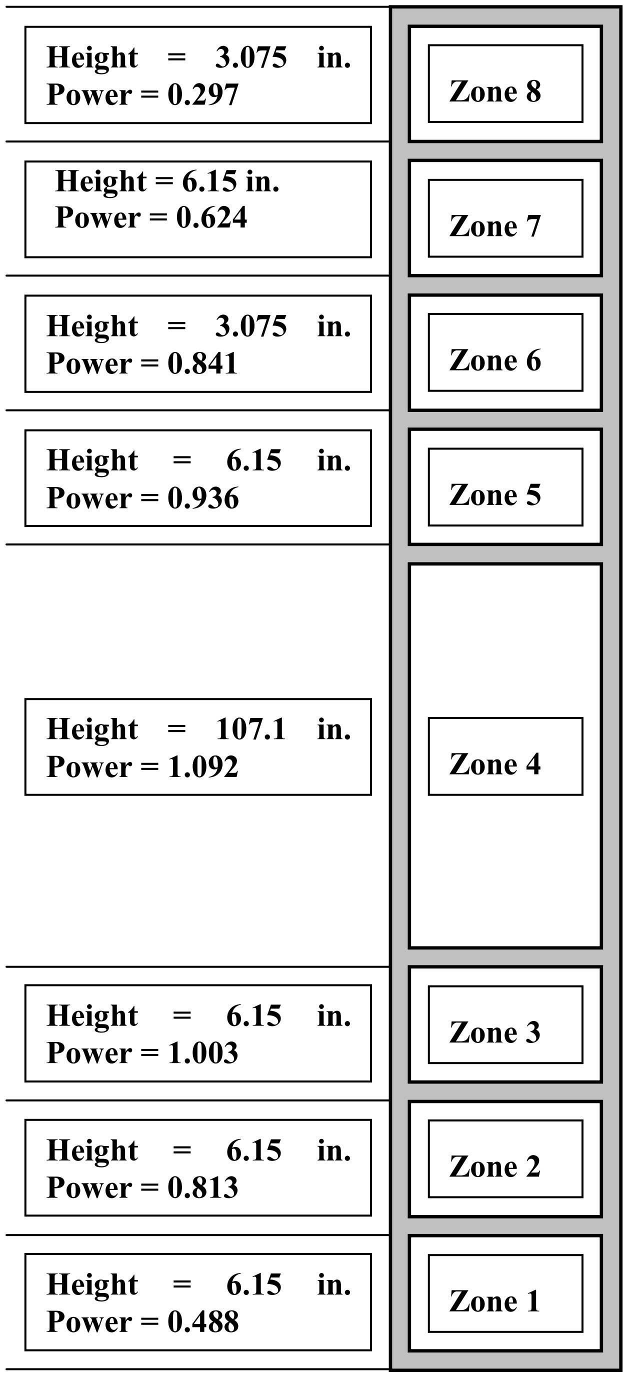


**Figure 3-4. KENO Output Plot of the "All Cell" Model**

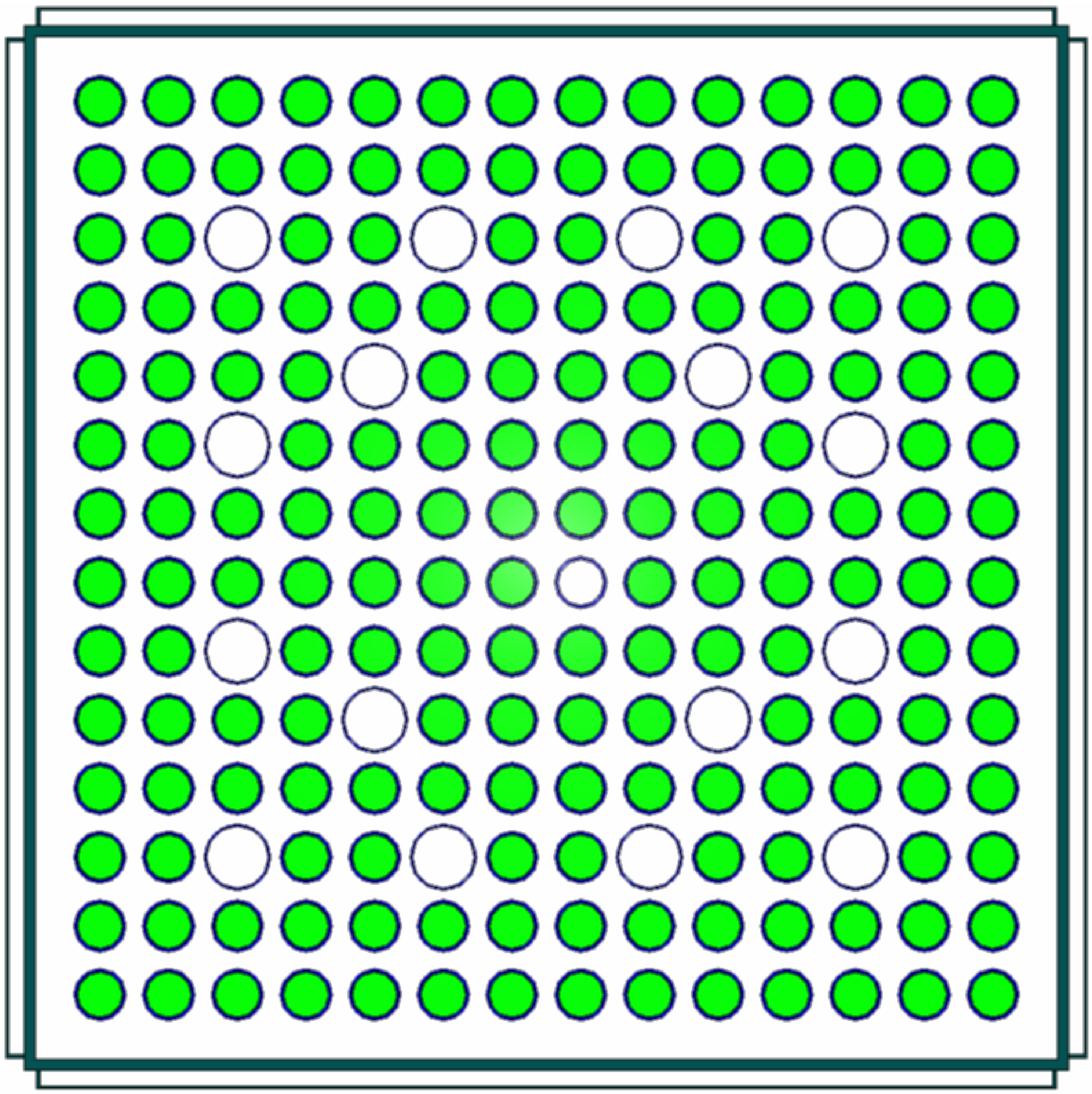


**Figure 3-5. KENO Output Plot of the "3x3" Storage Model**

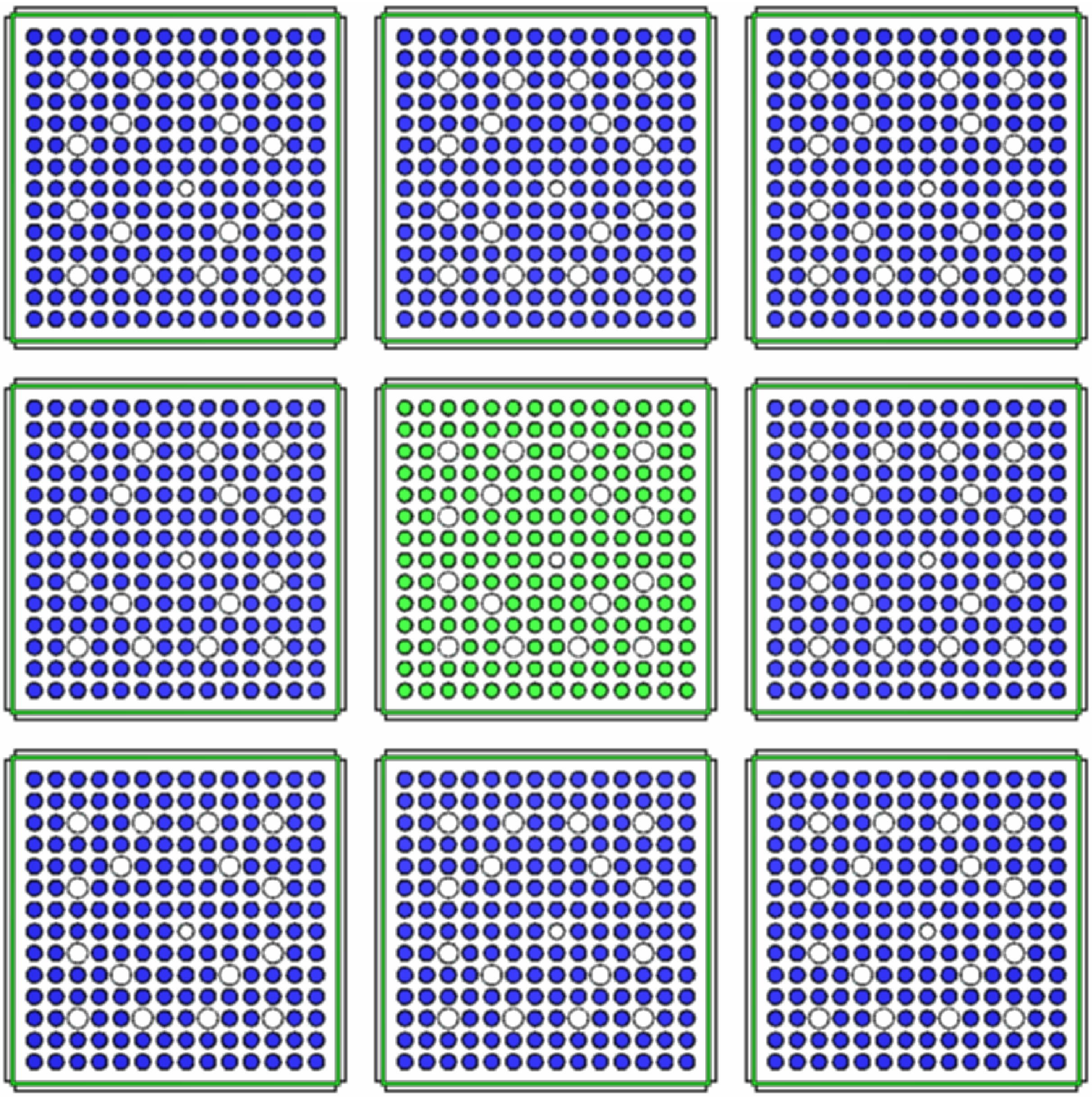
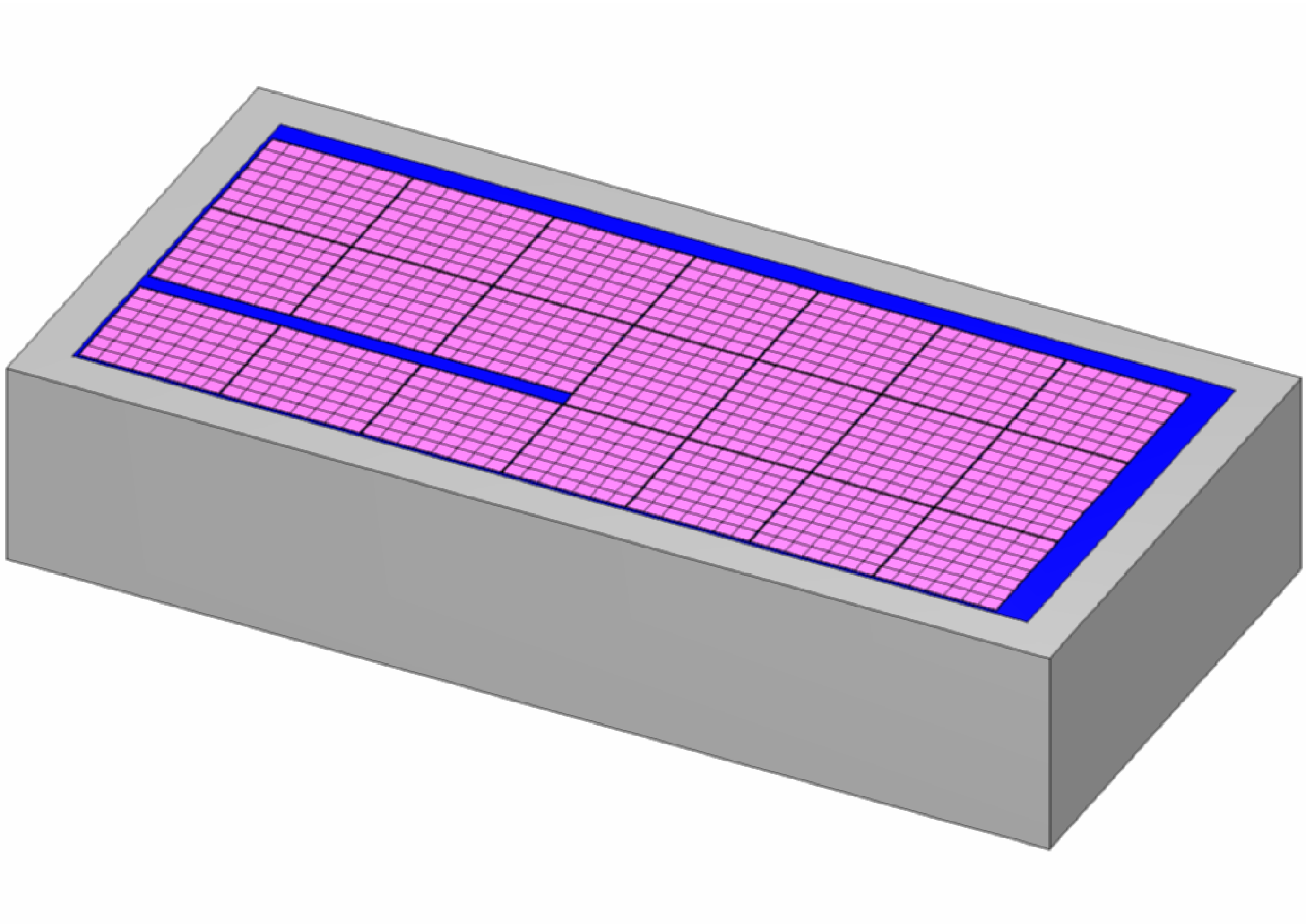
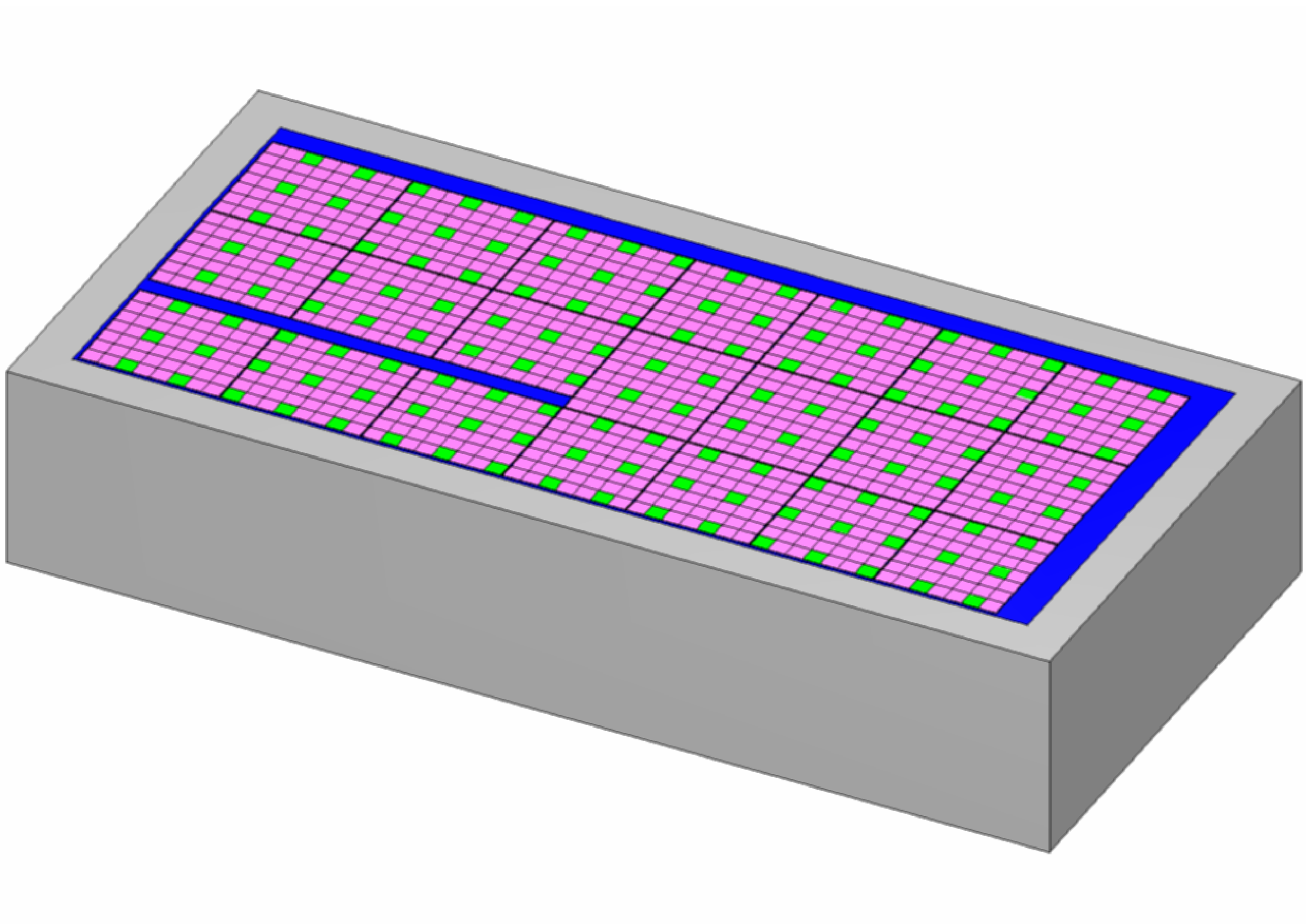


**Figure 3-6. KENO Output Plot of the "All-Cell" Spent Fuel Pool Model**



Note: Rack locations appear violet, pool water appears blue and pool wall appears gray in color.

**Figure 3-7. KENO Output Plot of the "3x3" Spent Fuel Pool Model**



Note: High enrichment rack locations appear green, low enrichment rack locations appear violet, pool water appears blue and pool wall appears gray in color.

# 4.0 Summary of Results

The following sections contain the criticality analysis results for the Prairie Island Units 1 & 2 spent fuel pool with soluble boron credit.

## 4.1 Allowable Storage Configurations and Interfaces

Figure 4-1 displays the allowable assembly arrangements for the All Cell storage configuration. The All Cell storage configuration will be employed to store depleted fuel assemblies that meet the requirements of Figure 4-5 or no fuel assembly. The assembly burnup versus initial enrichment limits should be calculated with the second degree polynomial given in Table 4-1 based upon the appropriate decay time.

Figure 4-2 displays the allowable assembly arrangements for the unshimmed “3x3” storage configuration. The unshimmed “3x3” storage configuration will be employed to store depleted fuel assemblies which meet the requirements of Figure 4-6 or no fuel assembly. The center storage cell will be employed to store fresh fuel assemblies of enrichment values of up to and including 4.95 w/o $^{235}U$ or no fuel assembly. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the third degree polynomial given in Table 4-2 based upon the appropriate decay time.

Figure 4-3 displays the allowable assembly arrangements for the $Gd_2O_3$ shimmed “3x3” storage configuration. The $Gd_2O_3$ shimmed “3x3” storage configuration will be employed to store depleted fuel assemblies which meet the requirements of Figure 4-7 or no fuel assembly. The center storage cell will be employed to store fresh fuel assemblies with a minimum of 4 $Gd_2O_3$ shimmed fuel rods. The minimum concentration of $Gd_2O_3$ is equal to 4.0 w/o. The maximum enrichment for unshimmed fresh fuel rods is equal to 4.95 w/o $^{235}U$. The maximum enrichment for $Gd_2O_3$ shimmed fresh fuel rods is equal to 4.0 w/o $^{235}U$. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the third degree polynomial given in Table 4-3 based upon the appropriate decay time.

The allowable interface between the storage configurations is displayed in Figure 4-4. Note that a row of empty storage cells at the interface may be used to separate the configurations. Also, it is acceptable to replace an assembly with an empty cell.

Note that a Failed Fuel Pin Basket (FFPB), a fully loaded Consolidation Rod Storage Basket (CRSB) with up to two (2) fuel rods missing, or a partially loaded CRSB with a maximum of 18 fuel rods may be substituted for any assembly in either the “All-Cell” or “3x3” fuel assembly storage configurations.

The FFPB is employed to store up to 16 fresh fuel rods (in a 4x4 array) with a maximum enrichment less than or equal to 4.95 w/o $^{235}U$ with no credit for burnup.

The fully loaded CRSC is a container designed to accommodate all of the fuel rods from two assemblies and fit into a single storage location. Note that up to two fuel rods may be missing from the container. The current burnup versus initial enrichment tech spec limits for the fully loaded CRSC are still valid.

A partially loaded CRSC may contain up to 18 fresh fuel rods with a maximum enrichment less than or equal to 4.95 w/o $^{235}U$ with no credit for burnup.

## 4.2 Burnup Credit

Figure 4-5 displays the assembly burnup versus initial enrichment storage curve for the All Cell storage configuration. The All Cell storage requirements are tabulated in Table 4-1 for 0, 5, 10, 15, and 20 years of decay time. The assembly burnup versus initial enrichment limits should be calculated with the second degree polynomial given in Table 4-1 based upon the appropriate decay time.

Figure 4-6 displays the burnup versus enrichment storage curve for the unshimmed “3x3” storage configuration. The unshimmed “3x3” storage requirements are tabulated in Table 4-2 for 0, 5, 10, 15, and 20 years of decay time. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the second degree polynomial given in Table 4-2 based upon the appropriate decay time.

Figure 4-7 displays the burnup versus enrichment storage curve for the shimmed “3x3” storage configuration. The shimmed “3x3” storage requirements are tabulated in Table 4-3 for 0, 5, 10, 15, and 20 years of decay time. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the second degree polynomial given in Table 4-3 based upon the appropriate decay time.

## 4.3 Total Soluble Boron Requirement

The total soluble boron (sum of all three components) required to maintain the $k_{eff}$ value (including all biases and uncertainties, without the adjustment for $^{10}B$) less than or equal to 0.95 is determined to be 722 ppm for a $^{10}B$ atom percent equal to 19.9. The soluble boron concentration required for a $^{10}B$ atom percent equal to 19.7 is 730 ppm. The recommended minimum boron level is 730 ppm and is sufficient to accommodate all the design requirements.

**Table 4-1. Fuel Assembly Burnup versus Initial Enrichment for the "All-Cell" Storage Configuration**

| Initial Enrichment (w/o $^{235}U$) | Burnup (MWD/MTU) | | | | |
|---|---|---|---|---|---|
| | Decay Time (years) | | | | |
| | 0 | 5 | 10 | 15 | 20 |
| 1.8 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| 3.0 | 13459.3 | 12715.2 | 12307.9 | 11989.1 | 11758.8 |
| 4.0 | 24542.7 | 23423.8 | 22555.1 | 21998.4 | 21526.0 |
| 5.0 | 35698.3 | 34082.9 | 32754.5 | 31937.0 | 31235.6 |

The 2nd degree polynomial that describes the 0 years decay time curve is as follows:

$$\boxed{\mathbf{BU = -41.86e^2 + 11{,}460.01e - 20{,}596.32}}$$

The 2nd degree polynomial that describes the 5 years decay time curve is as follows:

$$\boxed{\mathbf{BU = -5.19e^2 + 10{,}721.89e - 19{,}392.79}}$$

The 2nd degree polynomial that describes the 10 years decay time curve is as follows:

$$\boxed{\mathbf{BU = -34.77e^2 + 10{,}503.42e - 18{,}895.50}}$$

The 2nd degree polynomial that describes the 15 years decay time curve is as follows:

$$\boxed{\mathbf{BU = -32.36e^2 + 10{,}232.30e - 18{,}414.94}}$$

The 2nd degree polynomial that describes the 20 years decay time curve is as follows:

$$\boxed{\mathbf{BU = -41.66e^2 + 10{,}073.94e - 18{,}095.27}}$$

**Table 4-2. Fuel Assembly Burnup versus Initial Enrichment for the Unshimmed "3x3" Storage Configuration as a function of decay time**

| Initial Enrichment (w/o $^{235}$U) | Burnup (MWD/MTU) | | | | |
|---|---|---|---|---|---|
| | Decay Time (years) | | | | |
| | 0 | 5 | 10 | 15 | 20 |
| 1.3 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| 3.0 | 26549.6 | 24899.4 | 23376.5 | 22354.9 | 21420.7 |
| 4.0 | 38501.7 | 36140.4 | 34560.9 | 33044.3 | 32194.1 |
| 5.0 | 50281.8 | 46842.9 | 45644.9 | 43558.8 | 42671.7 |

The 3rd degree polynomial that describes the 0 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 343.67e^3 - 4210e^2 + 28706e - 30958}}$$

The 3rd degree polynomial that describes the 5 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 268.15e^3 - 3487.1e^2 + 25729e - 28143}}$$

The 3rd degree polynomial that describes the 10 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 243.34e^3 - 2970.3e^2 + 22973e - 25379}}$$

The 3rd degree polynomial that describes the 15 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 222.65e^3 - 2759.3e^2 + 21766e - 24122}}$$

The 3rd degree polynomial that describes the 20 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 142.91e^3 - 1862.8e^2 + 18525e - 21249}}$$

**Table 4-3. Fuel Assembly Burnup versus Initial Enrichment for the $Gd_2O_3$ Shimmed "3x3" Storage Configuration as a function of decay time**

| Initial Enrichment (w/o $^{235}U$) | Burnup (MWD/MTU) | | | | |
|---|---|---|---|---|---|
| | Decay Time (years) | | | | |
| | 0 | 5 | 10 | 15 | 20 |
| 1.46 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| 3.0 | 22119.5 | 20619.7 | 19840.6 | 18964.7 | 18512.6 |
| 4.0 | 33019.7 | 31255.7 | 29914.7 | 29070.0 | 28328.7 |
| 5.0 | 44022.3 | 42192.6 | 40271.4 | 39100.4 | 38187.8 |

The 3rd degree polynomial that describes the 0 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 399.61e^3 - 4744.1e^2 + 29324e - 33943}}$$

The 3rd degree polynomial that describes the 5 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 348.72e^3 - 4034.2e^2 + 25973e - 30406}}$$

The 3rd degree polynomial that describes the 10 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 352.36e^3 - 4087.1e^2 + 25646e - 29828}}$$

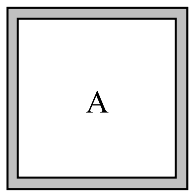
The 3rd degree polynomial that describes the 15 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 235.15e^3 - 2859.2e^2 + 21419e - 25909}}$$

The 3rd degree polynomial that describes the 20 years decay time curve is as follows:

$$\boxed{\mathbf{BU = 251.32e^3 - 2994.3\, e^2 + 21477e - 25756}}$$

**Figure 4-1. Allowable Fuel Assembly Combinations for the "All-Cell" Storage Configuration**



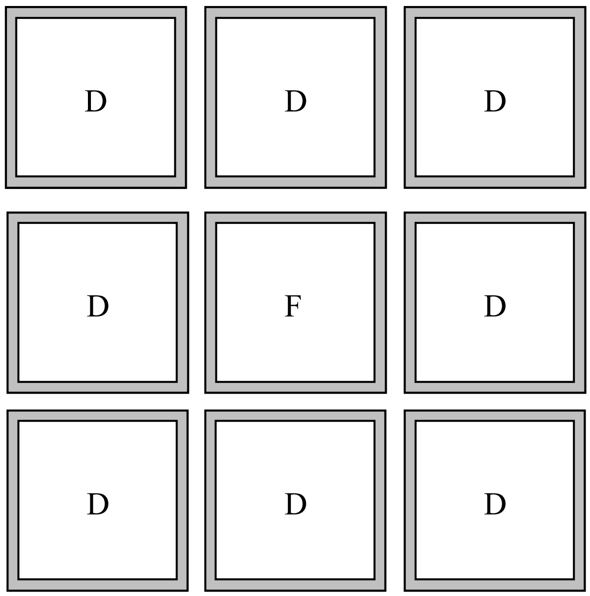

"A" represents

- Depleted fuel assembly which meets the requirements of Figure 4-5.

or

- An empty location.

**Figure 4-2. Allowable Fuel Assembly Combinations for the Unshimmed "3x3" Storage Configuration**




"D" represents

- Depleted fuel assembly which meets the requirements of Figure 4-6.

<u>or</u>

- An empty location.

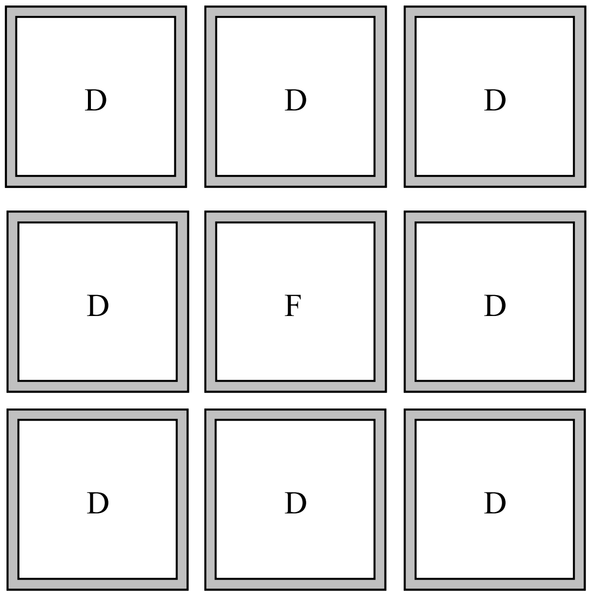
"F" represents

- Fresh unshimmed fuel assembly of enrichment values up to and including 4.95 $^{w}/_{o}$ $^{235}U$.

<u>or</u>

- An empty location.

**Figure 4-3. Allowable Fuel Assembly Combinations for the $Gd_2O_3$ Shimmed "3x3" Storage Configuration**




"D" represents

- Depleted fuel assembly which meets the requirements of Figure 4-7.

<u>or</u>

- An empty location.

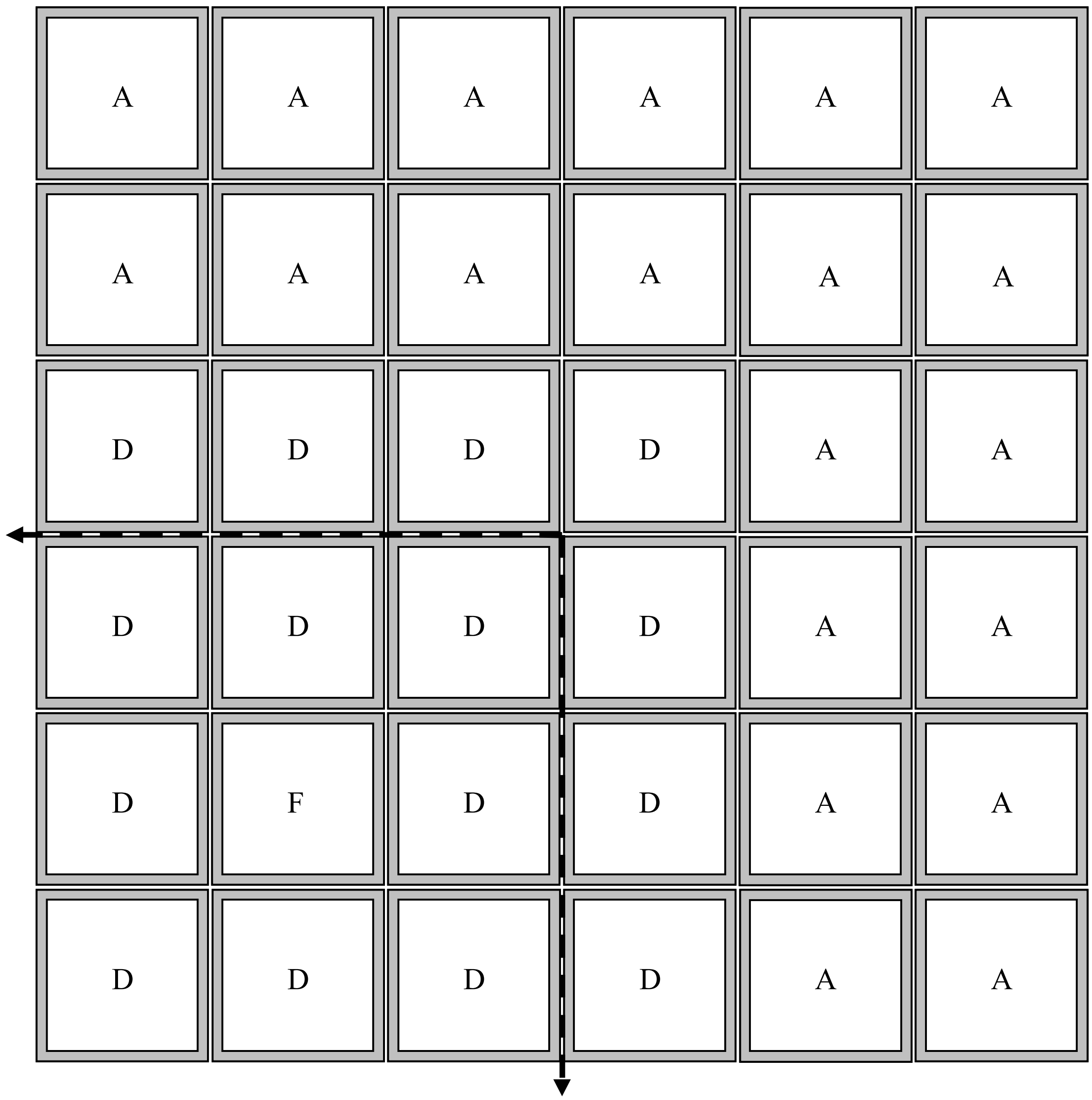
"F" represents

- Fresh $Gd_2O_3$ shimmed fuel assembly. Minimum of 4 shimmed fuel rods with a minimum of 4 w/o $Gd_2O_3$. Maximum enrichment for unshimmed rods is 4.95 $^w/_o$ $^{235}U$. Maximum enrichment for shimmed rods is 4.0 $^w/_o$ $^{235}U$.

<u>or</u>

- An empty location.

**Figure 4-4. Boundary Between the "3x3" and "All-Cell" Storage Configurations**




A = "All-Cell" Storage Location per Figure 4-1
F and D = "Fresh" and "Depleted" Fuel Assemblies per Figure 4-2 or 4-3, as appropriate

Notes:

1) A row of empty cells can be used at the interface to separate the configurations.

2) It is acceptable to remove an assembly from any storage location.

**Figure 4-5. Prairie Island Units 1 & 2 Assembly Burnup Requirements for the "All-Cell" Storage Configuration**

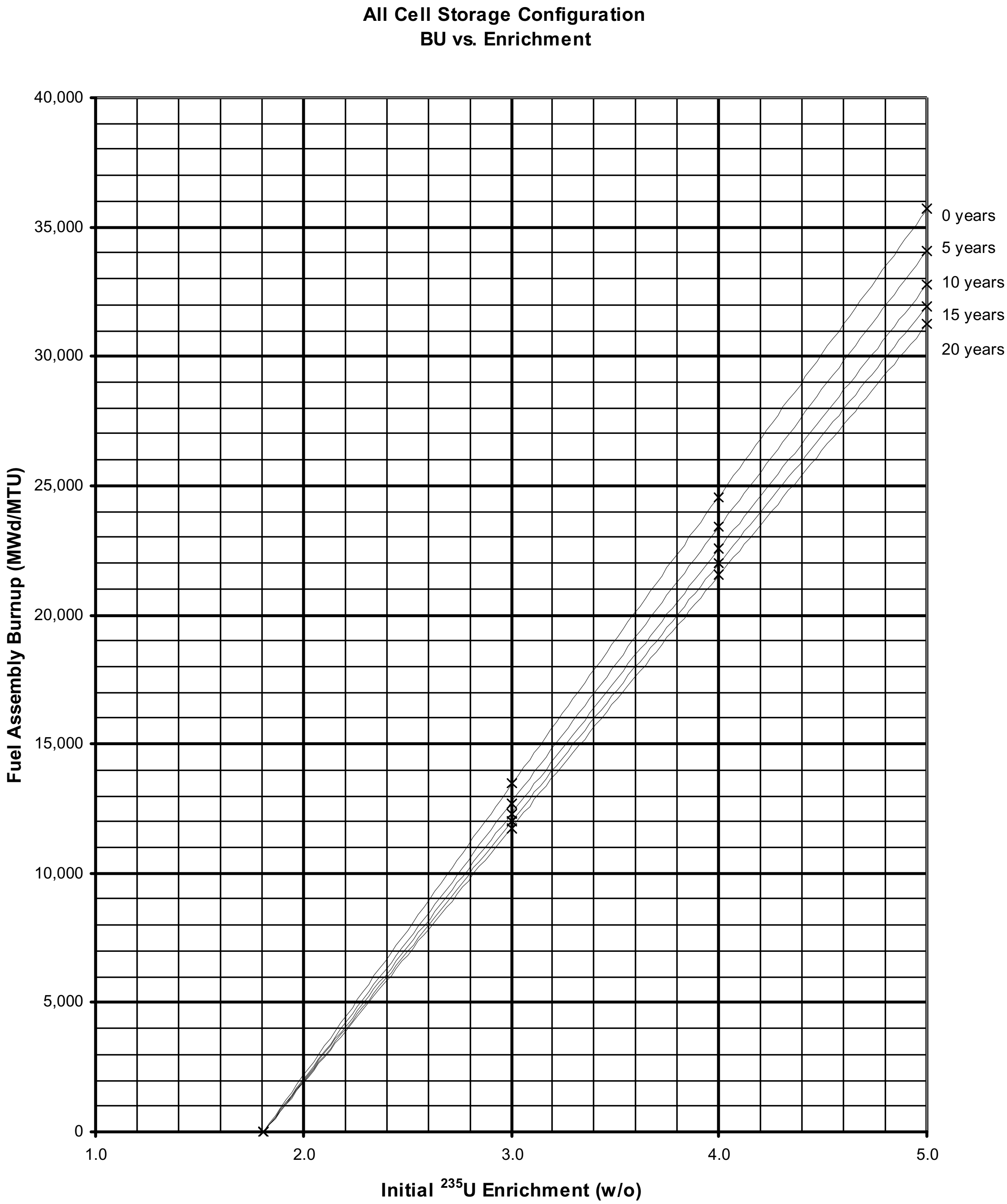


**Figure 4-6. Assembly Burnup Requirements for the Peripheral Fuel Assemblies in the Unshimmed "3x3" Storage Configuration**

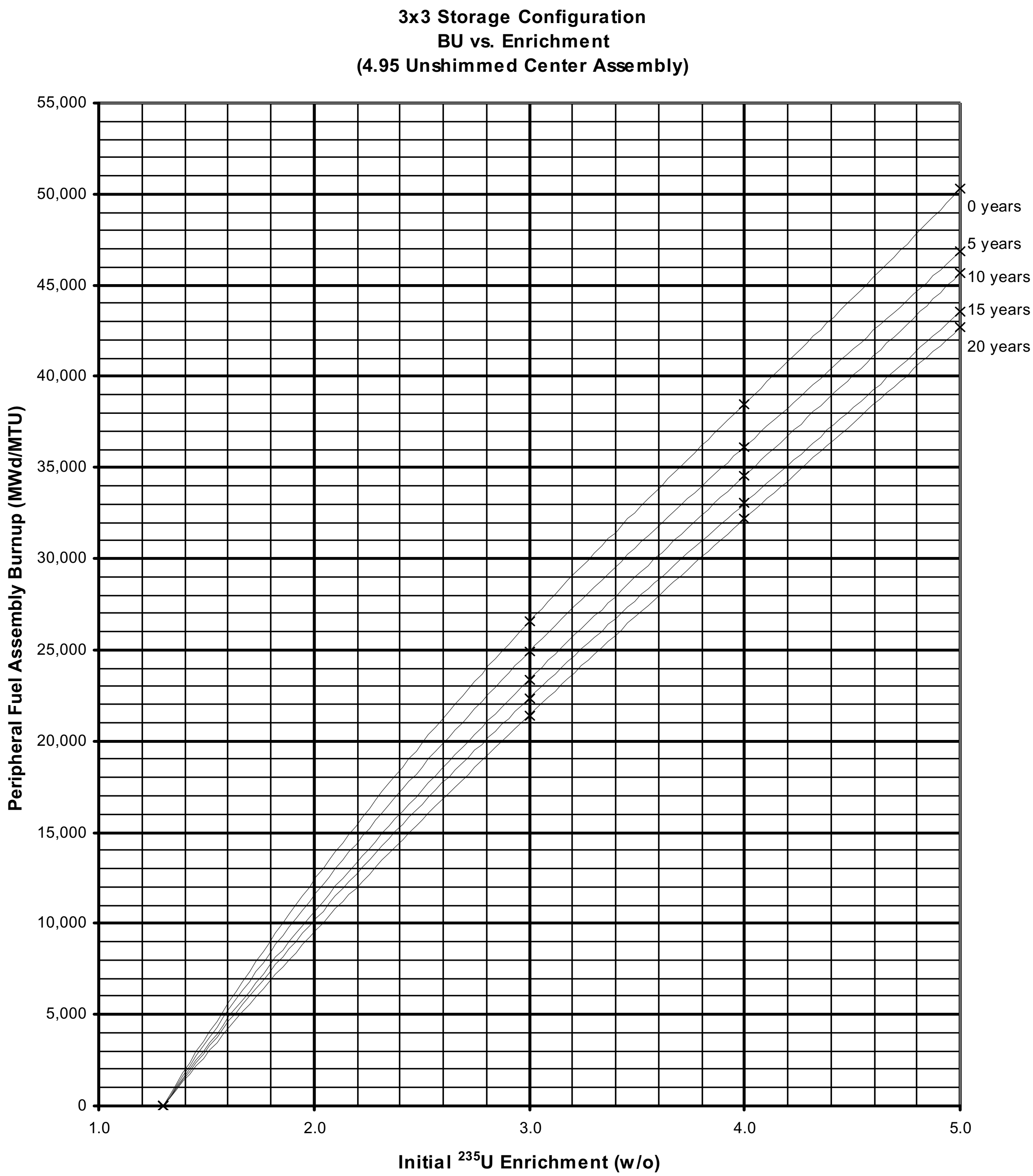
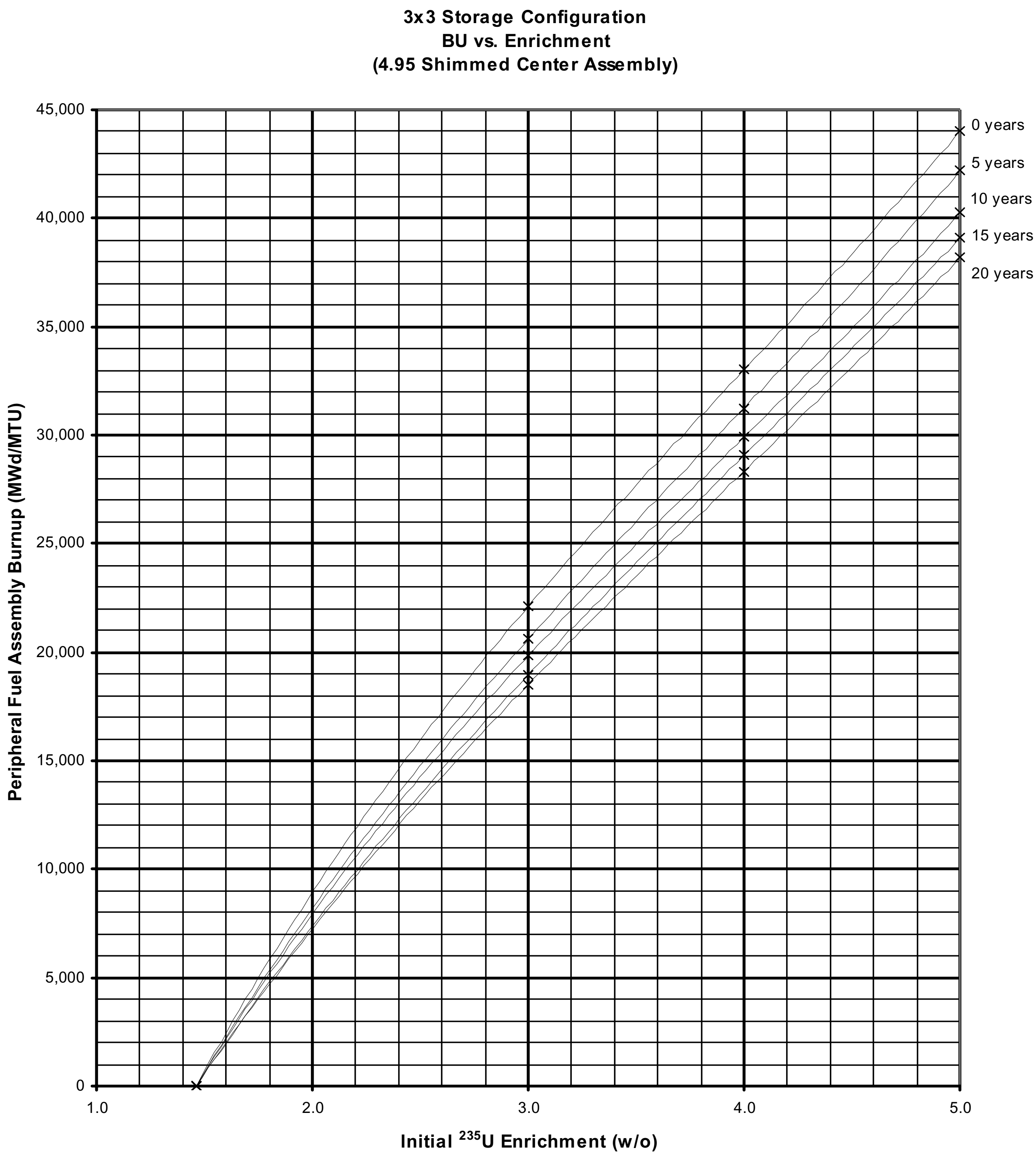


**Figure 4-7. Assembly Burnup Requirements for the Peripheral Fuel Assemblies in the Shimmed "3x3" Storage Configuration**



# 5.0 Computer Codes Used In Calculation

Table 5-1

Summary of Computer Codes Used in Calculation

| Code No. | Code Name | Code Version | Verified and Configured per EP-310 or EP-313? (Yes/No) or Configuration Control Reference | Basis (or reference) that supports use of code in current calculation | Outstanding Category A Error? (Yes/No). If Yes, how acceptable? |
|---|---|---|---|---|---|
| 1 | SCALE-PC | 4.3 | Yes | QC-1 | No |

## 6.0 References


1. Nuclear Fuel Engineering Procedure EP-302, "Documentation and Verification of Design Analysis".

2. Letter, G. S. Vissing (NRC) to R. C. Mecredy (RGE), "R. E. Ginna Nuclear Power Plant – Amendment Re-Revision to the Storage Configuration Requirements within the Existing Storage Racking and Taking Credit for a Limited Amount of Soluble Boron", December 7, 2000.

3. Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)", October 25, 1996.

4. AEC GDC proposed Criterion 66 – Prevention of Fuel Storage Criticality, July 10, 1967.

5. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", February 1998.

6. "SCALE 4.3- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers", NUREG/CR-200; distributed by the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1993.

7. "DIT: Discrete Integral Transport Assembly Design Code", CE-CES-11, Revision 4-P, April 1994.

8. M. N. Baldwin, et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel; Summary Report", BAW-1484-7, July 1979.

9. S. R. Bierman and E.D. Clayton, "Critical Experiments with Subcritical Clusters of 2.35 Wt% $^{235}U$ Enriched UO2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6", NUREG/CR-1547, PNL-3314, July 1980.

10. S.R. Bierman and E.D. Clayton, "Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 Wt% U-Enriched UO2 Rods in Water with Steel Reflecting Walls", Nuclear Technology, Vol. 54, pg. 131, August 1981.

11. International Handbook of Evaluated Criticality Safety Benchmark Experiments, Nuclear Energy Agency and Organization for Economic Cooperation and Development.

12. W. Marshall, et al., "Criticality Safety Criteria", TANS Vol. 35, pg. 278, 1980.

13. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans", SCR-607, Sandia Corporation Monograph, March 1963.

14. "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages", DOE/RW-0472 Rev. 1, May 1997.

15. P. A. Narayanan, "Isotopic Number Densities for Discharged Westinghouse 17X17 Fuel Assemblies," A-GEN-FE-0118, Revision 0, September 26, 2000.

16. Parrington, Josef R., et al, "Nuclides and Isotopes: Chart of the Nuclides", Fifteenth Edition, General Electric Co. and KAPL, Inc., 1996.

17. "Implementation of Zirlo™ Cladding Material in CE Nuclear Power Fuel Assembly Designs", CENPD-404-P, Rev. 0, January 2001.

18. "Methodology Manual for Cross Section Tableset Generation", CE-CES-124, Revision 2-P, June 1994.

19. P.F. O'Donnell, et al, "R.E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit", RGE-09-0009, Revision 0, November 1999.

20. J.R. Lesko, et al., "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", CAA-97-042, Revision 0, February 1997.

# A. Supporting Documentation

NMC
Committed to Nuclear Excellence

**Nuclear Management Company, LLC**
**Prairie Island Nuclear Generating Plant**
1717 Wakonade Dr. East • Welch MN 55089

Paul O'Donnell
Westinghouse Electric Company
102 Addison Road
Windsor, CT. 06095

May 28, 2003

Enclosed are the drawings for the Prairie Island spent fuel pool and racks. Prairie Island has 2 shared pools for both units. The smaller pool contains spent fuel racks and space for fuel casks, and the larger pool contains spent fuel racks. The smaller pool has space for 9 individual racks. 4 of these are considered temporary racks and are removable to make space for casks. With the 4 temporary installed, we have space for 1582 assemblies. All of our analyses include the 4 racks and 1582 assemblies.

The drawings included are:

NF-39213 General arrangement of the pool
NF-90044 This shows the arrangement of the various racks in the pool, including spacing. There are 2 styles of racks at Prairie Island, 7x7 and 7x8 arrays.
NF-90046 through NF-90052 These show the construction of the 7x7 and 7x8 racks.

If you have any questions about these please call me at 651/388-1121, ext 4819.

Jon Kapitz
Superintendent, Nuclear Engineering

## B. Computer Run Log Summary

| Run No | Table 5-1 Code No's | Computer Run Description | Machine Name | Run Date/ Time | File Type | EDMS File Name or File Location |
|---|---|---|---|---|---|---|
| **All Cell Storage, Bias & Uncertainty** | | | | | | |
| | 0 | Nominal | PC | 02/24/2003 | output | uv8y4abw.cdf |
| | 0 | Increase in $^{235}U$ Enrichment | PC | 02/24/2003 | output | uv8y4ef5.cdf |
| | 0 | Increase in Stack Density | PC | 02/24/2003 | output | uv8y4ihu.cdf |
| | 0 | Decrease in Cell Pitch | PC | 02/24/2003 | output | uv8y4mk4.cdf |
| | 0 | Decrease in Rack Thickness | PC | 02/24/2003 | output | uv8y4qns.cdf |
| | 0 | Decrease in Rack ID | PC | 02/25/2003 | output | uv8y4vpz.cdf |
| | 0 | Off-Center Assembly Positioning | PC | 02/24/2003 | output | uv8y51tq.cdf |
| **All Cell Storage, Fresh Equivalent Enrichment** | | | | | | |
| | 0 | 1.73 w/o $^{235}U$ 0 MWD/MTU | PC | 02/24/2003 | output | uv8y55vx.cdf |
| | 0 | 5 w/o $^{235}U$ 55,000 MWD/MTU | PC | 02/24/2003 | output | uv8y6gng.cdf |
| **All Cell Storage, Distributed Burnup** | | | | | | |
| | 0 | 3 w/o $^{235}U$ 15,000 MWD/MTU | PC | 02/24/2003 | output | uv8y6kpq.cdf |
| | 0 | 3 w/o $^{235}U$ 25,000 MWD/MTU | PC | 02/24/2003 | output | uv8y6pte.cdf |
| | 0 | 3 w/o $^{235}U$ 35,000 MWD/MTU | PC | 02/24/2003 | output | uv8y6wx3.cdf |
| | 0 | 4 w/o $^{235}U$ 25,000 MWD/MTU | PC | 02/24/2003 | output | uv8y73zc.cdf |
| | 0 | 4 w/o $^{235}U$ 35,000 MWD/MTU | PC | 02/24/2003 | output | uv8y7933.cdf |
| | 0 | 4 w/o $^{235}U$ 45,000 MWD/MTU | PC | 02/24/2003 | output | uv8y7f5s.cdf |
| | 0 | 5 w/o $^{235}U$ 35,000 MWD/MTU | PC | 02/24/2003 | output | uv8y7j91.cdf |

| | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/24/2003 | output | uv8y7ob8.cdf |
| | 0 | 5 w/o $^{235}$U 55,000 MWD/MTU | PC | 02/24/2003 | output | uv8y7teu.cdf |

| Run No | Table 5-1 Code No's | Computer Run Description | Machine Name | Run Date/ Time | File Type | EDMS File Name or File Location |
|---|---|---|---|---|---|---|
| **3x3 Storage, Bias & Uncertainty** | | | | | | |
| | 0 | Nominal | PC | 02/25/2003 | output | uv8ycuoq.cdf |
| | 0 | Increase in $^{235}$U Enrichment | PC | 02/25/2003 | output | uv8yczs7.cdf |
| | 0 | Increase in Stack Density | PC | 02/24/2003 | output | uv8yd4vx.cdf |
| | 0 | Decrease in Cell Pitch | PC | 02/25/2003 | output | uv8yd8yr.cdf |
| | 0 | Decrease in Rack Thickness | PC | 02/25/2003 | output | uv8ydd2w.cdf |
| | 0 | Increase in Rack ID | PC | 02/25/2003 | output | uv8ydj5d.cdf |
| | 0 | Off-Center Assembly Positioning | PC | 02/25/2003 | output | uv8ydn9e.cdf |
| | 0 | 0.955 Stack Density | PC | 02/24/2003 | output | uv8ydtc3.cdf |
| | 0 | 0.935 Stack Density | PC | 02/24/2003 | output | uv8ydyg7.cdf |
| | 0 | 0.915 Stack Density | PC | 02/24/2003 | output | uv8ye2iq.cdf |
| **3x3 Storage, Fresh Equivalent Enrichment** | | | | | | |
| | 0 | 1.36 w/o $^{235}$U 0 MWD/MTU | PC | 02/25/2003 | output | uv8ye8mr.cdf |
| **3x3 Storage, Distributed Burnup** | | | | | | |
| | 0 | 3 w/o $^{235}$U 15,000 MWD/MTU | PC | 02/25/2003 | output | uv8yfihs.cdf |
| | 0 | 3 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/25/2003 | output | uv8yfnjz.cdf |
| | 0 | 3 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/25/2003 | output | uv8yfrnq.cdf |

| | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | 4 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/25/2003 | output | uv8yfwpx.cdf |
| | 0 | 4 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/25/2003 | output | uv8yg3v3.cdf |
| | 0 | 4 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/25/2003 | output | uv8yga1h.cdf |
| | 0 | 5 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/25/2003 | output | uv8yge3o.cdf |
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/25/2003 | output | uv8ygi7f.cdf |
| | 0 | 5 w/o $^{235}$U 55,000 MWD/MTU | PC | 02/25/2003 | output | uv8ygn9r.cdf |

| Run No | Table 5-1 Code No's | Computer Run Description | Machine Name | Run Date/ Time | File Type | EDMS File Name or File Location |
|---|---|---|---|---|---|---|
| **3x3 Storage, Distributed Burnup, 5 Year Plutonium Decay** | | | | | | |
| | 0 | 3 w/o $^{235}$U 15,000 MWD/MTU | PC | 02/25/2003 | output | uv8ylthf.cdf |
| | 0 | 3 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/25/2003 | output | uv8ylxjm.cdf |
| | 0 | 3 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/25/2003 | output | uv8ym3nd.cdf |
| | 0 | 4 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/25/2003 | output | uv8ym8r4.cdf |
| | 0 | 4 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/25/2003 | output | uv8ymdtd.cdf |
| | 0 | 4 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/25/2003 | output | uv8ymhx3.cdf |
| | 0 | 5 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/25/2003 | output | uv8ymm0m.cdf |
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/25/2003 | output | uv8yms31.cdf |
| | 0 | 5 w/o $^{235}$U 55,000 MWD/MTU | PC | 02/25/2003 | output | uv8ymy6s.cdf |

| | | | | | | |
|---|---|---|---|---|---|---|
| **3x3 Storage, Distributed Burnup, 10 Year Plutonium Decay** | | | | | | |
| | 0 | 3 w/o $^{235}$U 15,000 MWD/MTU | PC | 02/25/2003 | output | uv8yn4aj.cdf |

| | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | 3 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/26/2003 | output | uv8ynaea.cdf |
| | 0 | 3 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8ynhhm.cdf |
| | 0 | 4 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/26/2003 | output | uv8ynmk8.cdf |
| | 0 | 4 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8ynqnz.cdf |
| | 0 | 4 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/26/2003 | output | uv8ynwro.cdf |
| | 0 | 5 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8yo0tx.cdf |
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/26/2003 | output | uv8yo5xo.cdf |
| | 0 | 5 w/o $^{235}$U 55,000 MWD/MTU | PC | 02/26/2003 | output | uv8yo9zy.cdf |

| **3x3 Storage, Distributed Burnup, 15 Year Plutonium Decay** | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | 3 w/o $^{235}$U 15,000 MWD/MTU | PC | 02/26/2003 | output | uv8yod3p.cdf |
| | 0 | 3 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/26/2003 | output | uv8yoh3d.cdf |
| | 0 | 3 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8yor7v.cdf |
| | 0 | 4 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/26/2003 | output | uv8yowb1.cdf |
| | 0 | 4 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8yp2f1.cdf |
| | 0 | 4 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/26/2003 | output | uv8yp6hu.cdf |
| | 0 | 5 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8ypbl6.cdf |
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/26/2003 | output | uv8ypiou.cdf |
| | 0 | 5 w/o $^{235}$U 55,000 MWD/MTU | PC | 02/26/2003 | output | uv8ypnra.cdf |

| **3x3 Storage, Distributed Burnup, 20 Year Plutonium Decay** | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | 3 w/o $^{235}$U 15,000 MWD/MTU | PC | 02/26/2003 | output | uv8ypuvk.cdf |
| | 0 | 3 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/26/2003 | output | uv8ypz06.cdf |

| | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | 3 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8yq33a.cdf |
| | 0 | 4 w/o $^{235}$U 25,000 MWD/MTU | PC | 02/26/2003 | output | uv8yq868.cdf |
| | 0 | 4 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8yqe9z.cdf |
| | 0 | 4 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/26/2003 | output | uv8yqldq.cdf |
| | 0 | 5 w/o $^{235}$U 35,000 MWD/MTU | PC | 02/26/2003 | output | uv8yqqhh.cdf |
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU | PC | 02/26/2003 | output | uv8yqwkq.cdf |
| | 0 | 5 w/o $^{235}$U 55,000 MWD/MTU | PC | 02/26/2003 | output | uv8yr0nf.cdf |

| Run No | Table 5-1 Code No's | Computer Run Description | Machine Name | Run Date/Time | File Type | EDMS File Name or File Location |
|---|---|---|---|---|---|---|
| **Spent Fuel Pool** | | | | | | |
| | 0 | Nominal case | PC | 03/15/2003 | output | uv8lv10i.cdf |
| **Soluble Boron** | | | | | | |
| | 0 | 5 w/o $^{235}$U 45,000 MWD/MTU, Distributed | PC | 03/15/2003 | output | uv8lv52q.cdf |
| | 0 | 200 ppm | PC | 03/15/2003 | output | uv8lv94x.cdf |
| | 0 | 400 ppm | PC | 03/15/2003 | output | uv8lvd74.cdf |
| | 0 | 600 ppm | PC | 03/15/2003 | output | uv8lvi9b.cdf |
| | 0 | 800 ppm | PC | 03/15/2003 | output | uv8lvmbi.cdf |
| | 0 | 1000 ppm | PC | 03/15/2003 | output | uv8lvrdp.cdf |

| Accident Cases | | | | | | |
|---|---|---|---|---|---|---|
| | 0 | Misloaded fresh fuel assembly into burnup storage rack location. | PC | 03/15/2003 | output | uv8lvwfw.cdf |
| | 0 | Misloaded fresh fuel assembly between All Cell storage racks and pool wall. | PC | 03/15/2003 | output | uv8lw0i3.cdf |
| | 0 | Intramodule water gap reduction due to seismic event. | PC | 03/15/2003 | output | uv8lw5ka.cdf |
| | 0 | Limiting accident with $SBC_{Total}$ | PC | 03/15/2003 | output | uv8lw9mh.cdf |

# C. Calculation Note Methodology Checklist
(Completed By Author)

| No | Self Review Topic | Yes | No | N/A |
|---|---|---|---|---|
| 1 | Is the subject and/or purpose of the calculation clearly stated? | | | |
| 2 | Are the required inputs and their sources provided and determined to be appropriate for the current calculation? Do the references contain sufficient information to clearly identify the source and facilitate its retrieval, including documents not maintained as quality records by Westinghouse? | | | |
| 3 | Are the assumptions clearly identified and justified? | | | |
| 4a | Are the methods clearly identified? | | | |
| 4b | If non-standard methods and/or computer codes (not described in a design manual or, in the absence of a design manual, not previously used in a licensing application for the applicable plant) were used, has a review been performed to determine whether there is a change to licensed methodology, and for analysis performed in support of USNRC plants is a form EPF-125-1, completed and attached? | | | |
| 4c | Were interim procedures required and appropriately used? | | | |
| 5 | Are the units of measurement clearly identified? | | | |
| 6 | Have the limits of applicability been identified? | | | |
| 7 | Are the results of literature searches, if conducted or other background data provided? | | | |
| 8 | Are all the pages sequentially numbered, and calculation note number and revision number listed on each page? | | | |
| 9 | Has the required computer calculation information been provided? Are all computer calculation outputs necessary to demonstrate that the objective of the analysis was accomplished, identified and included in the calculation note? | | | |
| 10a | Were the computer codes(s) used under Configuration Control? | | | |
| 10b | If computer codes were not controlled and previously validated and this analysis is used for a USNRC licensed plant, is a form, EPF-125-1 completed and attached? | | | |
| 11 | Were the computer codes(s) used applicable for modeling the physical and/or computational problem(s) contained in this calc note? | | | |
| 12 | Are the results and conclusions clearly stated? | | | |
| 13 | Are open items properly identified? | | | |
| 14 | Are all hand-annotated changes to the calculation note initialed and dated by author and verifier? Has a single line been drawn through any changes with the original information remaining legible ? | | | |
| 15 | Are all references clearly identified in the calculation note and in any attachment? | | | |
| 16 | Have applicable Application-Specific Verification Checklist(s) been completed and attached. (Optional, attach if used) | | | |

If 'NO' to any of the above, provide cross-reference to justification or provide additional explanation below or on subsequent pages.

## D. Verification Methodology Checklist

(Completed By Verifier(s))

| Verification Method (One or more must be completed by each verifier | | Check If Performed |
|---|---|---|
| 1. | Independent review of document. (Briefly explain method of review below or attach) | |
| 2. | Verification performed by alternative calculations as indicated below (1): | |
| | a. Comparison to a sufficient number of simplified calculations which give persuasive support to the original analysis. | |
| | b. Comparison to an analysis by an alternate verified method. | |
| | c. Comparison to a similar verified design or calculation. | |
| | d. Comparison to test results. | |
| | e. Comparison to measured and documented plant data for a comparable design. | |
| | f. Comparison to published data and correlation's confirmed by experience in the industry. | |
| 3. | Other (Describe under Additional Details of Verifier's Review below) | |

(1) For independent verification accomplished by comparisons with results of one or more alternate calculations or processes, the comparison should be referenced, shown below, or attached to the checklist.

| Complete by checking the appropriate box | | Yes | No | N/A |
|---|---|---|---|---|
| 1. | Does the Verifier concur with the author's entries in Checklists A and B, including "N/A" entries and the justification for any "No" responses? | | | |
| 2. | Does the Verifier concur with the author's entries in any Application-Specific checklists? | | | |

**Verification:** The verifier's approval indicates that all comments or necessary corrections identified during the review of this document have been incorporated as required. This document has been verified using the method(s) described above. For multiple verifiers, indicated appropriate methods(s) by initials.

**<u>Additional Details of Verifier's Review</u>**

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# E. Additional Verifier's Comments

Use this page to document any additional checks performed, along with any relevant technical comments not addressed in the calculation note and the author's response. Note if a response by the author is not required, i.e., "no response required." Author's responses or written justification should be documented here.

| No | Verifier's Comments | Author's Response (Indicate if not required) | Verifier Concurs? (Yes/No) |
|---|---|---|---|
| 1 | Page 1 – number of pages is incorrect. | Fixed | Yes |
| 2 | General – make sure you are using the correct template (NFE Version 4-1). | Criticality documents employ unique format | Yes |
| 3 | General – remove "draft" from all headings. | Fixed. | Yes |
| 4 | General – need to justify using the isotopic concentrations generated for Ginna (1520 MWt) for Prairie Island (1650 MWt). | Minor difference on 2D/3D reactivity bias due to assembly specific power with 10 % of nominal value. | Yes |
| 5 | Section 2.3 – the ID for the SS sheathing is noted as 8.7 in. In the KENO models, it seems to be different. Please verify, and address the importance if different. | Verified that dimension is correctly modeled. | Yes |
| 6 | Section 3.2 – Where are the cases for the scoping calculations? The reviewer has to trust that they were done, and that the STD and OFA types were indeed limiting. | Scoping calculation results were not saved for this analysis. Note that the design basis fuel assemblies assumed for these configurations are very typical. | Yes |
| 7 | Section 3.4 – Include the tolerance on the Gad concentration. | Added a description for gad. tolerance. | Yes |
| 8 | Section 3.6.2 – How is the 0.01736 value computed? | The delta k-effective for 5 % burnup uncertainty was conservatively calculated. | Yes |
| 9 | Table 3.1 – information on cell pitch and instrumentation tube is missing. Also, the OFA cladding used was Zirc-4. Is this conservative (as opposed to Zirlo)? | Cell pitch and instrumentation tube data added to table. Note concerning zr-4 modeling as conservative also added. | Yes |
| 10 | Table 3.4 – there are errors with some of the keffs listed. How is the burnup uncertainty computed? The methodology uncertainty appears incorrect, please verify. The statistical sum of uncertainties also appears incorrect.<br><br>These questions apply to Tables 3.5 and 3.6 also. | K-effective values listed correctly. One of the uncertainties was updated. Sum of uncertainties was updated. Sum of uncertainties and biases is correct. Burnup uncertainty equal to 5 % of maximum burnup for configuration.<br><br>Table 3-5 and 3-6 also updated. | Yes |

| | | | |
|---|---|---|---|
| 11 | Table 3.5 – the nominal case enrichment is incorrect.<br>This also applies to Table 3.6. | The value for the nominal enrichment was labeled incorrectly, updated to 1.2 w/o.<br>The value for the nominal enrichment was labeled incorrectly, updated to 1.5 w/o. | Yes |
| 12 | Table 3.6 – the sum of uncertainties and biases is incorrect, given multiple errors in the table. | The sum of the uncertainties and biases is correct. Sum of the uncertainties was corrected. Nominal case k-effective updated. Methodology uncertainty is correct. Several of the uncertainties updated. | Yes |
| 13 | Table 3.13 – incorrect numbers in table. | K-effective results for unshimmed configuration updated. | Yes |
| 14 | Table 3.14 – incorrect numbers in table. | All values have been updated. Polynomial is correct. | Yes |
| 15 | Figure 3.1 – text box is empty. | Text box was fixed. | Yes |
| 16 | Section 4.1 – statements are made regarding the use of FFPBs or CRSBs. Where are the analyses supporting these statements? | Customer requested that we review analysis of records concerning these items and include statements in this document concerning their limits. | Yes |
| 17 | Table 4.1 – how are the polynomials generated? Why use 2nd order for some, 3rd order for others? Also, the values in the tables can not be reproduced from the polynomials.<br><br>These issues apply to Tables 4.2 and 4.3 also. | Resolved with reviewer. | Yes |
| 18 | Figure 4.4 – how is this configuration determined? This is never discussed, or related to the one interface case that was executed. | We ran a specific KENO calculation for the interface. Section 3.5.3 updated to reflect the interface modeled in KENO. | Yes |
| 19 | KENO cases – for the tolerance cases, why is 4.95 w/o used as the base case for the density and enrichment tolerance, as opposed to the “nominal” enrichments? | The nominal enrichment case would produce an unrealistically high results. The nominal enrichment is associated with discharged fuel assemblies. | Yes |
| 20 | KENO cases – for the 3x3_50_45k_20 and 3x3_50_55k_20, the O-16 concentration is missing in zone 8. | The effect of O-16 on moderation is extremely small. | Yes |
| 21 | KENO cases – the Gad tolerance case gives a lower keff value than the nominal case. Perhaps the tolerance is not large enough, or something is wrong. | The gad tolerance case is within one sigma of the nominal case. This indicates that the gad is “black” to neutrons. | Yes |

| | | | |
|---|---|---|---|
| 22 | KENO cases – please provide the relationship between ppm and the values used in the sfp decks. | We model ppm as grams B-10 per million grams water. The B-10 is in $H_3BO_3$ (Boric Acid). Therefore, $H_3BO_3$ density supplied to KENO equal to a constant * water density * ppm . | Yes |
| 23 | KENO cases – please provide the relationship between Gad concentration (4.0 w/o) and the compositions provided in the decks. | The $Gd_2O_3$ content of $UO_2$- $Gd_2O_3$ = 4.0 % Therefore, gad isotopes modeled with overall density times 0.04 and isotopic concentration. | Yes |
| 24 | Feel free to consider editorial markings throughout document. | NONE required | Yes |
| | | | |