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3. System	4. Document Identifier
CSNF Disposal Container	CAL-DSU-NU-000005 REV 00A

5. Title
BWR Axial Profile

6. Group
Criticality

7. Document Status Designation
☒ Preliminary ☐ Final ☐ Cancelled

8. Notes/Comments
Performing a AP-2.14Q *Document Review* is not necessary for this product since the results of this calculation do not impact any other organization.

Attachments	Total Number of Pages
I. Cosine Profile	8
II. Parametric Study	6
III. Testing Criteria	6
IV. Poison Selection	6
V. Checking Process	6
VI. CD File Description	8
VII. Compact Disc Attachment	1 of 1 CDs

RECORD OF REVISIONS

9. No.	10. Reason For Revision	11. Total # of Pgs.	12. Last Pg. #	13. Originator (Print/Sign/Date)	14. Checker (Print/Sign/Date)	15. QER (Print/Sign/Date)	16. Approved/Accepted (Print/Sign)	17. Date
00A	Initial issue.	96	VI-8	Jason Huffer 9/27/04 Alan Wells 9/27/04	Jeff Ryman 9/27/04 Alan Wells 9/27/04	Darrell Svalstad 9/27/04	William Hutchins	9/28/04

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1. PURPOSE

The purpose of this calculation is to develop axial profiles for estimating the axial variation in burnup of a boiling water reactor (BWR) assembly spent nuclear fuel (SNF) given the average burnup of an assembly. A discharged fuel assembly typically exhibits higher burnup in the center and lower burnup at the ends of the assembly. Criticality safety analyses taking credit for SNF burnup must account for axially varying burnup relative to calculations based on uniformly distributed assembly average burnup due to the under-burned tips. Thus, accounting for axially varying burnup in criticality analyses is also referred to as accounting for the "end effect" reactivity. The magnitude of the reactivity change due to "end effect" is dependent on the initial assembly enrichment, the assembly average burnup, and the particular axial profile characterizing the burnup distribution. The set of bounding axial profiles should incorporate multiple BWR core designs and provide statistical confidence (95 percent confidence that 95 percent of the population is bound by the profile) that end nodes are conservatively represented. The profiles should also conserve the overall burnup of the fuel assembly. More background on BWR axial profiles is provided in Attachment I.

ANSI/ANS-8.17-1984, "*Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel outside Reactors*", Section 4.10 specifically allows analysis and verification of the exposure history of each fuel unit for fuel burnup credit, with the stated condition that consideration shall be given to the axial distribution of burnup.

The applicability of this calculation is limited to BWRs of similar design to the three reactors that have compilations of Commercial Reactor Criticality data (CRCs): Grand Gulf Unit 1 (Punatar 2001), Quad Cities Unit 2 (CRWMS M&O 1999a) and LaSalle Unit 1 (CRWMS M&O 1999b). Since the design of LaSalle Unit 1 incorporates sophisticated design features such as partial length rods, this calculation is applicable to all commercial BWR SNF generated in the United States.

This report is an engineering calculation supporting the development of analyses to be used for license application of the monitored geologic repository, and was performed under OCRWM procedure AP-3.12Q, *Design Calculations and Analyses*. This calculation is subject to *Quality Assurance Requirements and Description* (DOE 2004), per the activity evaluation under *Technical Work Plan for: Criticality Department Work Packages ACRM01 and NSN002* (BSC 2004).

2. METHOD

The calculation method to determine the set of bounding burnup profiles uses three compilations of CRCs (CRWMS M&O 1999a; CRWMS M&O 1999b; and Punatar 2001). The reactor data sets are presented in 25- and 24-node format. The three references compile the reactor CRC data into 10-node representations, which are used to determine the set of bounding profiles. The bounding profiles are a mathematical tool to allow criticality safety analysts to assign conservative burnup values to axial nodes of SNF from the provided average burnup value per assembly.

As fuel is burned in a reactor, its burnup becomes distributed axially. The profile of this axial distribution attains a flattened cosine shape with time, although the exact profile can vary significantly with operating history and other effects unique to the individual reactor. Axial profiles from multiple reactors were used as representative of BWR fuel assemblies. The development of the profiles and reactor operating parameters are discussed in Section 5.

A set of isotopic concentrations is calculated for each axial section of the assembly. Axially dependent concentrations for each profile are used to calculate the k_{eff} for a flooded 44 BWR waste package (WP) containing a full set of identical assemblies.

The k_{eff} calculations take credit for burnup by using a subset of the total isotopes. The list of 29 isotopes for use with commercial SNF is known as the principal isotope list. The principal isotopes (*Principal Isotope Selection Report,* CRWMS M&O 1998a) are shown in Table 3-3 of *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003).

The depletion calculations are performed with the SAS2H control module of the SCALE 4.4A code system (SCALE V4.4A, STN: 10129-4.4A-00, CRWMS M&O 2000c). The k_{eff} calculations are performed with MCNP V4B2LV (MCNP V4B2LV, CSCI: 30033 V4B2LV, CRWMS M&O, 1998c). Based on fuel assembly design, power history, and operating data for the specific assemblies, a computational representation was developed for use with SAS2H and MCNP. The SAS2H module is used to perform a fuel depletion analysis using operating history parameters to predict the isotopic concentrations in localized areas of assembly pins. The isotopic concentrations predicted by the SAS2H module are then used as material input to MCNP to generate k_{eff} in a flooded 44-BWR WP.

3. ASSUMPTIONS

3.1 USE OF ALUMINUM

Assumption: It is assumed that using the aluminum (Al) material cross-section for zinc (Zn) in the MCNP cases has a negligible impact on the results of criticality calculations.

Rationale: The basis for this assumption is that the neutronic characteristics for Zn and Al are sufficiently similar. The Zn neutron cross-section libraries were not available for MCNP. In addition, the Zn material that is substituted only appears in Aluminum 6061 in trace amounts (BSC 2003a, Section 3).

Confirmation Status: This assumption does not need to be confirmed due to the small Zn neutron cross-section.

Use in the Calculation: Section 5.3, *Profile Demonstration,* and 5.7 *LaSalle Reactor Test* both utilizing depletion calculations with MCNP representations of 44-BWR waste packages utilizing the Al material. (BSC 2003a, Section 3)

3.2 TOTAL FUEL MASS

Assumption: It is assumed that during the fuel irradiation process of power production, the total fuel mass does not significantly change and the oxygen content of the original UO_2 fuel remains constant.

Rationale: This is because the rod casing contains the fuel throughout the process, and the oxygen does not absorb neutrons and is not a fission product.

Confirmation Status: This assumption does not need to be confirmed because it is the most conservative possibility.

Use in the Calculation: Section 5.3, *Profile Demonstration.*

3.3 BWR MODERATOR TEMPERATURE

Assumption: It is assumed that during the fuel irradiation process of power production, the moderator is at Core Inlet (Ci) temperature at the first node (center of Node 1 implies 2% of the core active fuel height). The temperature reaches the design core saturation temperature (Cs) by the 8^{th} node (center of Node 8 implies 78% of the core active fuel height).

Rationale: Since there is only a small temperature gradient (9° C) to spread over the core height, the assumption is a reasonable approximation.

Confirmation Status: This assumption does not need to be confirmed because it is conservative to treat the top of the core at the Cs temperature. Furthermore, the difference between the Ci temperature and Cs is only nine degrees centigrade in the available Grand Gulf data, which is an insignificant temperature gradient for neutronics cases.

Use in the Calculation: Section 5.5, *Other Temperatures*, utilizes the Cs at the 8^{th} node.

4. USE OF COMPUTER SOFTWARE

4.1 SAS2H

The SAS2H control module of the baselined SCALE (SCALE, V4.4A, STN: 10129-4.4A-00) modular code system was used to perform the fuel assembly depletion calculations required for this evaluation. The software specifications are as follows:

- Program Name: SAS2H of the SCALE Modular Code System
- Version/Revision Number: Version 4.4a; Ref: CRWMS M&O 2000c
- Status/Operating System: Qualified/HP-UX B.10.20
- Software Tracking Number (STN) Number: 10129-4.4A-00
- Computer Type: Hewlett Packard (HP) 9000 Series Workstations
- Computer Processing Unit (CPU) number: 700887

The input and output files for the various SAS2H calculations were documented electronically in a compact disk (CD), Attachment VII, to this calculation as described in Sections 5 and 8, such that an independent repetition of the software use may be performed. The SAS2H software used was:

1. Appropriate for the application of commercial fuel assembly depletion, and

2. Used only within the range of validation as documented in *Users Manual for SCALE-4.4A* (CRWMS M&O 2000a) and *Validation Test Report (VTR) for SCALE-4.4A* (CRWMS M&O 2000b), and

3. Obtained from Software Configuration Management in accordance with appropriate procedures.

4.2 MCNP

The baselined MCNP code (MCNP V4B2LV, CSCI: 30033 V4B2LV) was used to calculate the effective neutron multiplication factor (k_{eff}) for the various spent nuclear fuel compositions. The software specifications are as follows:

- Program Name: MCNP
- Version/Revision Number: Version 4B2LV; Ref: CRWMS M&O 1998c
- Status/Operating System: Qualified/HP-UX B.10.20
- Computer Software Configuration Item Number: 30033 V4B2LV
- Computer Type: HP 9000 Series Workstations
- CPU number: 700887

The input and output files for the various MCNP calculations are documented electronically in a CD, Attachment VII, to this calculation file as described in Sections 5 and 8, such that an independent repetition of the software use may be performed. The MCNP software used was:

1. Appropriate for the application of k_{eff} calculations,

2. Used only within the range of validation as documented throughout *MCNP-A General Monte Carlo N-Particle Transport Code* (Briesmeister 1997) and *Software Qualification Report for MCNP Version 4B2, A General Monte Carlo N-Particle Transport Code* (CRWMS M&O 1998b), and

3. Obtained from Software Configuration Management in accordance with appropriate procedures.

5. CALCULATION

The determination of the bounding profiles as a function of burnup group was performed for a set of BWRs by sorting profiles in the data set by burnup. The profiles were grouped into specified burnup groups, comparing the values of burnup for each axial node in the profile, and selecting the lowest values observed for the three nodes at each end of the active fuel rod.

For unknown distributions with greater than 58 data points, the 95 percent lower limit (95 percent confidence that 95 percent of the unsampled data is above the lowest point) is determined by the lowest value in the set. For larger numbers of data points, the 2nd- or 3rd-lowest number can represent the 95 percent lower limit. For consistency, the lowest value was used for each case (Natrella 1963, 2-5.4.2). Some of the individual reactor data sets did have less than 58 data points; however, in every case, the totals for the burnup groups are greater than 58 and are presented in Table 1.

The input data consisted of burnup values for the 25- or 24-nodes of each CRC and the 10-node scheme lengths of the rods, which varied for each reactor set. The following procedure was used to determine the burnup group profiles.

Many of the nodes in the 10-node scheme were compressions of several nodes in the 25- or 24-node CRC schemes as follows:

$$NBV_i = Average\ (CRC_{j=j1\text{-}j3}) \quad \text{(Eq. 1)}$$

where

NBV_i = Node Burnup Value of each individual node in the 10-node scheme

$CRC_{j=j1\text{-}j3}$ = Node burnup values in the published 25-node CRC scheme that are to be compressed. From one up to three (J=1, 2 or 3) of the 25-nodes are compressed together for a single node in the 10-node scheme.

The weight of each node in the 10-node scheme is the node burnup value (NBV_i) times the fraction of the total rod length the node represents.

$$W_i = NBV_i \times (Node\ Length_i / Total\ Node\ Length) \quad \text{(Eq. 2)}$$

Average burnup is the sum of the weights:

$$Average\ Burnup = \sum_{i=1}^{10} W_i \quad \text{(Eq. 3)}$$

The weighted burnup is a dimensionless value for each node, found by dividing the node burnup by the average burnup.

$$WBV_i = NBV_i / \text{Average Burnup} \quad \text{(Eq. 4)}$$

Normalizing the weighted burnup values (NWBV) give an adjusted node burnup value:

$$NWBV_i = \frac{WBV_i}{\sum_i WBV_i} \quad \text{(Eq. 5)}$$

This process generates a listing of average burnup values and the 10 normalized node burnups for each CRC available. The list is sorted by the average burnup values, allowing the data set to be split into groups of 10-14.9, 15-19.9, 20-24.9, 25-29.9, and 30-35 GWd/MTU.

To determine a bounding burnup profile, the six end nodes (1, 2, 3, 8, 9, and 10) are set to the minimum $T_{95/95}$ confidence level (95 percent confidence that 95 percent of the data is above the chosen node value). The $T_{95/95}$ confidence level is determined by Table A-31 in *Experimental Statistics* (Natrella 1963).

In equation form the profile terms are determined by:

$$P_{i=1\text{-}3,8\text{-}10} = T_{95/95} \text{ value of } (NWBV_{i=1\text{-}3,8\text{-}10}) \quad \text{(Eq. 6a)}$$

where

$NWBV_i$ = Set of normalized weighted node burnup values calculated for each grouping
P_i = Profile Value representing the set of node (N_i) values in a given grouping

The burnup values for the four interior nodes are derived from the total average value for the nodes less the value used in the six end nodes and renormalized such that the average burnup for the group was conserved. The interior four nodes are scaled up to conserve the overall burnup of the assembly.

$$P_{i=4\text{-}7} = \frac{\sum_{i=1}^{10}(\overline{N_i}) - \sum_{i=1}^{3} P_i - \sum_{i=8}^{10} P_i}{4} \quad \text{(Eq. 6b)}$$

where

$\overline{N_i}$ = The averaged value of all the normalized nodes in the grouping for a particular N_i.

Profiles for all of the available CRC data sets were compiled by using a weighted average. Table 1 is a summary of the data available, with average burnups presented with weighted burnups.

Table 1. Computation Summary

	10-14.9		15-19.9		20-24.9		25-29.9		30-34.9		35-40	
	Average Burnup	Count	Average Burnup	Count	Average Burnup	Count	Average Burnup	Count	Average Burnup	Count	Average Burnup	Count
Grand Gulf, A-E	13.5	224	17.3	303	22.2	282	27.7	352	31.9	199	36.4	41
Grand Gulf, F&G	12.2	85	16.5	50	22.6	61	28.0	47	31.9	132	36.9	48
Grand Gulf, K&N	14.7	8	16.5	72	N/A	0	N/A	0	N/A	0	N/A	0
G.G., H, J, L, M	13.7	94	16.6	183	22.7	72	28.1	77	30.2	9	N/A	0
Quad Cities, A-D	13.3	34	17.8	36	23.1	28	27.7	49	32.9	164	36.8	110
Quad Cities, E-M	12.7	111	17.6	112	22.4	93	27.6	56	31.9	56	N/A	0
LaSalle	11.9	65	17.0	173	22.4	97	27.1	94	32.6	73	37.4	88
Wt. Av./Total	13.1	621	17.0	929	22.4	633	27.7	675	32.2	633	36.9	287

Source: Spreadsheet: GGTH-Node, tab: Profile-Comp

The profiles were compiled with a weighted average and presented in Table 2.

Table 2. BWR Profile Summary

Burnup Ranges	Node 1 (bottom)	Node 2	Node 3	Nodes 4-7	Node 8	Node 9	Node 10 (top)
10-14.9	0.0196	0.1003	0.1310	0.1461	0.0879	0.0561	0.0207
15-19.9	0.0207	0.1035	0.1310	0.1448	0.0893	0.0565	0.0200
20-24.9	0.0214	0.1038	0.1305	0.1439	0.0901	0.0569	0.0218
25-29.9	0.0222	0.1026	0.1276	0.1430	0.0932	0.0606	0.0217
30-34.9	0.0224	0.1006	0.1242	0.1414	0.0978	0.0659	0.0233
35-40	0.0236	0.1001	0.1253	0.1409	0.1021	0.0641	0.0214

Source: Spreadsheet: GGTH-Node, tab: Profile-Comp

Node lengths had already been determined for each different reactor based on characteristics of the individual reactors (CRWMS M&O 1999a, p.15; CRWMS M&O 1999b, p. 12; and Punatar 2001, p. 3-4). The objective of this analysis is to incorporate multiple and varied reactors to develop a bounding set of BWR burnup profiles. The different axial length schemes were fitted to a common node length to produce a compiled profile set. The common node length was selected as the most occurring (mode) length of the individual node lengths, by percentage. The compilation of the seven node schemes and the most occurring, under the heading "Mode", is presented in Table 3.

Table 3. Seven Axial 10-Node Length Schemes

Node	LaSalle	Quad City Unit 2		Grand Gulf Unit 1				
		A-D	E-M	A-E	F & G	H, J, L & M	K & N	Mode[a]
1 (bottom)	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04
2	0.12	0.08	0.08	0.16	0.12	0.16	0.12	0.12
3	0.16	0.08	0.08	0.12	0.08	0.12	0.08	0.08
4	0.12	0.13	0.13	0.12	0.12	0.12	0.08	0.12
5	0.12	0.08	0.08	0.12	0.12	0.12	0.12	0.12
6	0.12	0.13	0.13	0.12	0.16	0.08	0.16	0.12
7	0.12	0.13	0.13	0.12	0.16	0.08	0.08	0.12
8	0.12	0.13	0.13	0.08	0.08	0.12	0.12	0.12
9	0.04	0.17	0.13	0.08	0.04	0.08	0.12	0.08
10 (top)	0.04	0.04	0.08	0.04	0.08	0.08	0.08	0.08
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Source: Spreadsheet: GGTH-Node, tab: Nodes
Note a: Appendix V-1 gives more detail on this determination.

The final step for usable axial profiles is to multiply the derived profiles, Table 2, by the selected node scheme (mode) percentages in Table 3. The resulting list is normalized. Each node in the normalized list has its node percentage divided back out. Table 4 presents the bounding axial profiles.

Table 4. BWR Bounding Axial Profiles

Burnup Ranges	Node 1 (bottom)	Node 2	Node 3	Nodes 4-7	Node 8	Node 9	Node 10 (top)
10-14.9	0.1782	0.9112	1.1901	1.3266	0.7978	0.5096	0.1883
15-19.9	0.1880	0.9405	1.1901	1.3154	0.8112	0.5136	0.1822
20-24.9	0.1950	0.9443	1.1869	1.3089	0.8201	0.5176	0.1982
25-29.9	0.2017	0.9339	1.1618	1.3024	0.8484	0.5517	0.1977
30-34.9	0.2042	0.9174	1.1328	1.2896	0.8918	0.6013	0.2122
35-40	0.2150	0.9130	1.1420	1.2843	0.9306	0.5845	0.1951

Source: Spreadsheet: GGTH-node, tab: Profile-comp

Bounding Axial Profiles are presented in Figure 1.

Given a specific burnup (power times days burned) for a BWR assembly, the power level for each node may be determined by multiplying the reported burnup by the set of profiles in Table 4. The node length can be found by multiplying the fraction from the mode column of Table 3 by the total active height of the fuel. The resulting burnup will provide a 95 percent confidence that 95 percent of the end burnup values would be conservative relative to the CRC data base information.

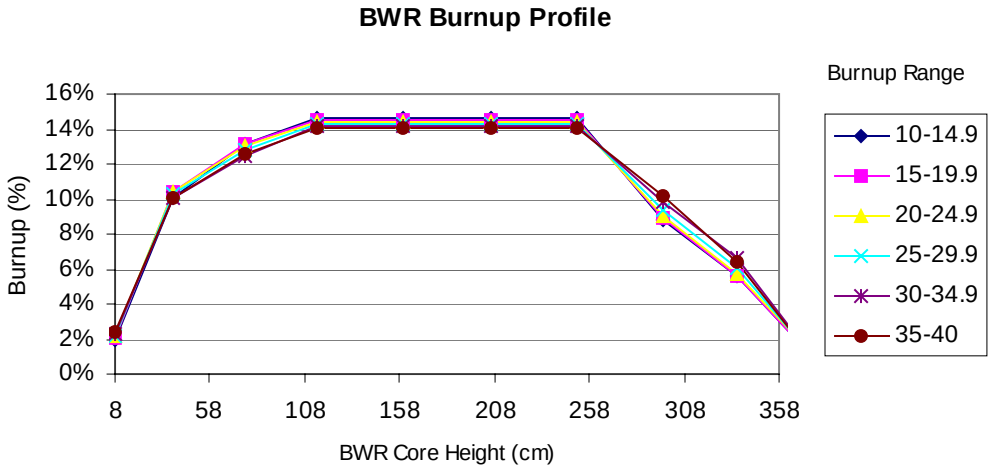

Figure 1. BWR Bounding Axial Profiles

Source: Spreadsheet: GGTH-node, tab: Profile-comp

5.1 MODERATOR DENSITIES AND TEMPERATURE PROFILES

The moderator densities and temperature profiles are also provided by the CRC data. To execute the SAS2H codes for the isotopic compositions required for the k_{eff} determination (MCNP runs), all three parameters are required.

Typical or average moderator densities and fuel temperature profiles were desired to demonstrate the above profiles. A parametric study was performed on the moderator density in Attachment II. Sensitivity studies of the affects of varying the fuel temperatures and control blade perturbations are beyond the scope of this study.

For the moderator densities, the average reported moderator densities were calculated for each reactor data set and the weighted average was computed as demonstrated for the 10-14.9 GWd/MTU data set shown in Table 5.

Table 5. Typical Moderator Density Profile

Burnup Range: 10-14.9	Average Burnup	Count	Node 1 (bottom)	Node 2	Node 3	Node 4	Node 5	Node 6	Node 7	Node 8	Node 9	Node 10 (top)
A-E	13.51	224	0.7649	0.7340	0.5486	0.4103	0.3246	0.2705	0.2347	0.2008	0.1911	0.1876
F & G	12.24	85	0.6995	0.6787	0.5336	0.3862	0.2810	0.2151	0.1715	0.1455	0.1496	0.1315
K & N	14.68	4	0.7367	0.7155	0.5684	0.4440	0.3444	0.2606	0.2156	0.1932	0.1688	0.1542
H, J, L, M	13.68	94	0.7539	0.7176	0.5259	0.3746	0.2838	0.2454	0.2235	0.1984	0.1972	0.1744
QC A-D	13.29	34	0.6815	0.6815	0.6159	0.4841	0.3898	0.3291	0.2793	0.2419	0.2127	0.2006
QC E-M	12.68	111	0.6781	0.6730	0.5993	0.4745	0.3804	0.3147	0.2610	0.2253	0.2014	0.1915
LaSalle	11.89	65	0.7396	0.7259	0.5931	0.4588	0.3802	0.3262	0.2889	0.2621	0.2505	0.2474
Wt. Average	13.04	617	0.7311	0.7090	0.5607	0.4225	0.3320	0.2760	0.2371	0.2059	0.1955	0.1854

Source: Spreadsheet: GGTH-Node, tab: Mod-comp

The sets of moderator density averages are presented in Table 6.

Table 6. Moderator Density Averages

Burnup Range	Node 1 (bottom)	Node 2	Node 3	Node 4	Node 5	Node 6	Node 7	Node 8	Node 9	Node 10 (top)
10-14.9	0.7311	0.7090	0.5607	0.4225	0.3320	0.2760	0.2371	0.2059	0.1955	0.1854
15-19.9	0.7388	0.7102	0.5449	0.4052	0.3181	0.2645	0.2290	0.1982	0.1883	0.1785
20-24.9	0.7381	0.7146	0.5667	0.4322	0.3434	0.2869	0.2461	0.2137	0.2026	0.1937
25-29.9	0.7432	0.7199	0.5711	0.4368	0.3477	0.2908	0.2502	0.2176	0.2058	0.1972
30-34.9	0.7221	0.7077	0.5945	0.4673	0.3749	0.3115	0.2631	0.2279	0.2121	0.2011
35-40	0.7294	0.7146	0.5940	0.4665	0.3798	0.3191	0.2737	0.2412	0.2273	0.2171

Source: Spreadsheet: GGTH-Node, tab: Mod-comp

Moderator density averages are shown in Figure 2.

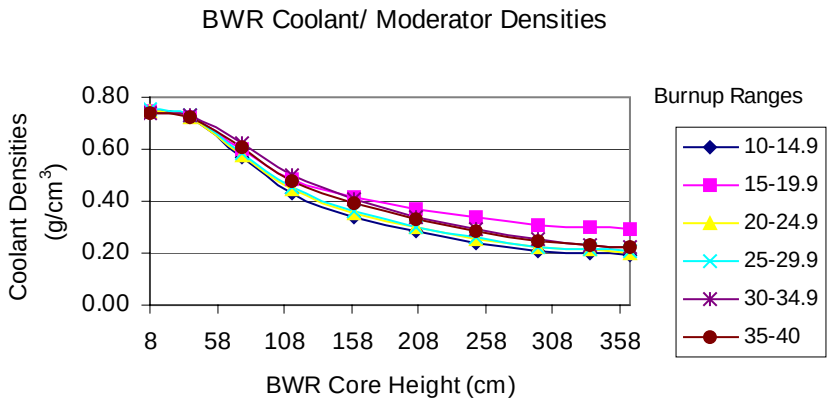

Figure 2. Moderator Density Profiles

Source: Spreadsheet: GGTH-Node, tab: Mod-comp

To match the derived burnup profiles; the moderator densities for nodes 4-7 were averaged together, as shown in Table 7.

Table 7. Moderator Density Profiles

Burnup Range	Node 1 (bottom)	Node 2	Node 3	Nodes 4-7	Node 8	Node 9	Node 10 (top)
10-14.9	0.7311	0.7090	0.5607	0.3169	0.2059	0.1955	0.1854
15-19.9	0.7388	0.7102	0.5449	0.3042	0.1982	0.1883	0.1785
20-24.9	0.7381	0.7146	0.5667	0.3272	0.2137	0.2026	0.1937
25-29.9	0.7432	0.7199	0.5711	0.3314	0.2176	0.2058	0.1972
30-34.9	0.7221	0.7077	0.5945	0.3542	0.2279	0.2121	0.2011
35-40	0.7294	0.7146	0.5940	0.3598	0.2412	0.2273	0.2171

Source: Spreadsheet: GGTH-Node, tab: Mod-comp

The moderator temperature is also a required input for SAS2H depletion code. However, BWR moderator temperature does not vary significantly, as more energy is absorbed by the moderator more boiling occurs, while the temperature remains relatively constant.

The fuel temperature profiles were developed by averaging the CRC data by burnup, as shown in Table 8.

Table 8. Typical Fuel Pellet Temperature (K) Average

Burnup Range: 10-14.9	Average Burnup	Count	Node1 (bottom)	Node 2	Node 3	Node 4	Node 5	Node 6	Node 7	Node 8	Node 9	Node 10 (top)
Grand Gulf	13.30	407	631	939	1006	1011	1006	984	954	892	791	629
Quad Cities	12.82	145	649	924	1001	1005	1021	1019	1006	968	840	642
LaSalle	11.89	65	657	1077	1176	1164	1143	1109	1023	875	662	621
Wt. Average	13.04	617	638	950	1023	1026	1024	1006	973	908	789	631

Source: Spreadsheet: GGTH-Node, tab: temp-comp

The set of fuel temperature averages is presented in Table 9.

Table 9. Fuel Pellet Temperature (K) Averages

Burnup Range	Node1 (bottom)	Node 2	Node 3	Node 4	Node 5	Node 6	Node 7	Node 8	Node 9	Node 10 (top)
10-14.9	638	950	1023	1026	1024	1006	973	908	789	631
15-19.9	640	940	997	999	989	968	934	865	756	623
20-24.9	632	870	926	940	937	918	885	832	739	620
25-29.9	629	831	877	900	909	902	882	835	744	623
30-34.9	613	738	766	785	797	798	788	762	701	610
35-40	615	732	758	784	806	823	827	793	705	616

Source: Spreadsheet: GGTH-Node, tab: temp-comp

Fuel temperature averages are presented in Figure 3.

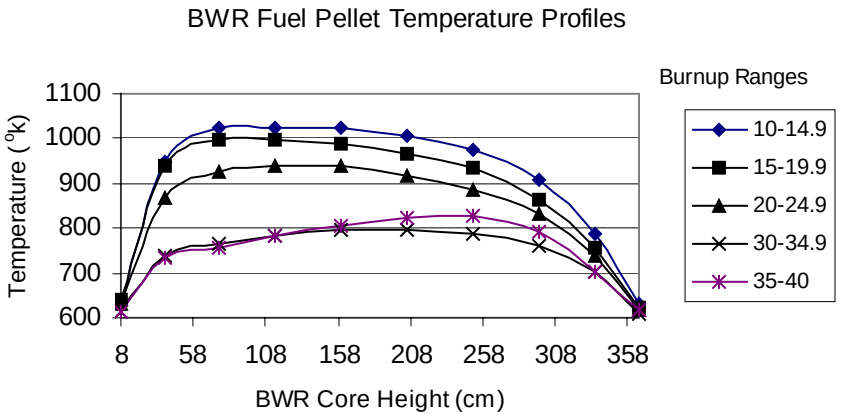

Figure 3. BWR Temperature Averages

Source: Spreadsheet: GGTH-Node, tab: temp-comp

It is of interest to note the distinctive temperature depression in the higher burnup cases, demonstrating the burnup of U-235 in the lower core.

To generate temperature profiles, nodes 4-7 were averaged together and presented in Table 10.

Table 10. Fuel Temperature (K) Profiles

Burnup Ranges	Node 1 (bottom)	Node 2	Node 3	Nodes 4-7	Node 8	Node 9	Node 10 (top)
10-14.9	638	950	1023	1007	908	789	631
15-19.9	640	940	997	972	865	756	623
20-24.9	632	870	926	920	832	739	620
25-29.9	629	831	877	898	835	744	623
30-34.9	613	738	766	792	762	701	610
35-40	615	732	758	810	793	705	616

Source: Spreadsheet: GGTH-Node, tab: temp-comp

5.2 SAS2H FUEL DEPLETION DESCRIPTION

The SAS2H control sequence accesses five calculational modules of the SCALE code system for performing fuel depletion and decay calculations. The five modules are BONAMI, NITAWL-II, XSDRNPM, COUPLE, and ORIGEN-S. Each of the modules has a specific purpose in the sequence to perform the fuel depletion and decay calculations. The following provides a brief description of what each module does with a more detailed description provided in *Users Manual for SCALE-4.4A* (CRWMS M&O 2000a).

- BONAMI – applies the Bondarenko method of resonance self-shielding to nuclides for which Bondarenko data are available.
- NITAWL-II – performs Nordheim resonance self-shielding corrections for nuclides that have resonance parameter data available.
- XSDRNPM – performs a one-dimensional neutron transport calculation in a specified geometry then computes cell-weighted cross sections.
- COUPLE – updates cross sections in an ORIGEN-S working nuclear data library with data from the cell-weighted cross section library obtained from the XSDRNPM calculation. Additionally, the weighting spectrum produced by XSDRNPM is applied to update all nuclides in the ORIGEN-S working library, which were not included in the XSDRNPM calculation.
- ORIGEN-S – performs point depletion, buildup, and decay calculations for the specified assembly irradiation history. ORIGEN-S can also be run as a stand-alone case to provide isotopic concentrations at various decay times.

The SAS2H control module uses ORIGEN-S to perform a point depletion calculation for the fuel assembly section described in the SAS2H input file. The ORIGEN-S module uses cell-weighted cross sections based on one-dimensional transport calculations performed by XSDRNPM. One-dimensional transport calculations are performed on two mock-ups, path A and path B, to calculate energy dependent spatial neutron flux distributions necessary to perform cross section cell-weighting calculations.

The path A model is simply a unit cell of the fuel assembly lattice containing a fuel rod. In the path A model, the fuel, clad, and moderator are modeled explicitly. The only modification required in developing the path A model is the conversion of the fuel assembly's square lattice unit cell perimeter to a radial perimeter conserving moderator volume within the unit cell (exterior to the fuel rod cladding). The SAS2H control module performs this modification automatically. A one-dimensional transport calculation is performed on the path A model for each energy group, and the spatial flux distributions for each energy group are used to calculate cell-weighted cross sections for the fuel.

The path B model is a larger representation of the assembly than the path A model. The path B model approximates spectral effects due to heterogeneity within the fuel assembly such as water gaps, burnable poison rods, control blades, or axial power shaping rods. The structure of the path B model is based on a uniform distribution of nonfuel lattice cells. In reality, most fuel assemblies do not have uniformly distributed nonfuel lattice cells. However, the approximation that the cells are uniform is considered acceptable within these calculations as documented in Section S2.2.3.1 of *Users Manual for SCALE-4.4A* (CRWMS M&O 2000a).

The path B model for the fuel assembly depletion calculations performed in this analysis includes an inner region composed of a representation of the nonfuel assembly lattice cell. A region containing the homogenization of the path A model surrounds the inner region in the path B model. A final region representing the moderator in the assembly-to-assembly spacing surrounds the homogenized region in the path B model. The size of each radial region that surrounds the inner region in the path B model is determined by conserving both the fuel-to-moderator mass ratio and the fuel-to-absorber (burnable poison) mass ratio in the corresponding section of the fuel assembly. The cell-weighted cross-sections from the path A model are applied to the homogenized region during the path B model transport calculations. New cell-weighted cross sections for each burnup group are then developed using the unit cell spatial flux distribution results from the path B model transport calculations. These cell-weighted cross sections are ultimately used in the point depletion calculations performed by ORIGEN-S to calculate the depleted fuel isotopic compositions in the corresponding fuel assembly. A detailed description of how SAS2H produces time-dependent cross sections is documented in Section S2.2.4 of *Users Manual for SCALE-4.4A* (CRWMS M&O 2000a).

5.3 PROFILE DEMONSTRATION

Calculation of Isotopic Bias and Uncertainty for BWR SNF (BSC 2003a) contains a compilation of four different BWRs complete with descriptions, material details, SAS2H input development and Monte Carlo inputs. To demonstrate use of the above profiles, one BWR (Cooper) was selected and the seven SAS2H inputs from the profiles were developed for each MCNP case. The cases were

selected to have started with 3, 4 and 5 wt. % U-235 initial fuel loading and 3.4 wt.% gadolinium loading in rods corresponding to the reactor data.

The burnup groups were selected in the center points of their ranges: 12.5, 17.5, 22.5, 27.5, 32.5 and 37.5 GWd/MTU. The six groups resulted in six burnup cases at three initial enrichments. Each case requires 18 MCNP runs with seven SAS2H depletion codes to support each MCNP code. This process resulted in 126 SAS2H cases required to perform the demonstration. With 29 isotopes collected per SAS2H case, it required 3,654 individual isotopic concentrations to recover, scale and transfer to the MCNP inputs.

For this demonstration the following power history was developed. Artificial time steps of 360 days followed by 60 days down time were selected and the 5-year cooling period approximated with 1,825 days. The burnup applied in each step of the fuel cycle was kept below the maximum burn of 8,000 (GWd/MTU) per cycle used in the bias calculation (BSC 2003a). Table 11 tabulates the demonstration burn. The average power level of 17.3612 MW/MTU is selected to keep the GWd/MTU below 8,000 for each cycle and the average burnup equals the 37.5 GWd/MTU.

The profile is then reproduced in Table 11, and each term is multiplied by the average rod burnup to give the power level for each node (an individual SAS2H run). The burnup column is in italics because it is not a direct SAS2H input; the burns required for each node SAS2H run are in bold because they change with each case. The moderator density and fuel temperature inputs also changed with each case per Tables 7 and 10.

Table 11. Demonstration Time Steps and Node Burns for 37.5 GWd/MTU

Power	**Days**	***Burnup***	**Down-time**				
17.3612	360	*6250.032*	60	Rod Average			
17.3612	360	*6250.032*	60				
17.3612	360	*6250.032*	60				
17.3612	360	*6250.032*	60				
17.3612	360	*6250.032*	60				
17.3612	360	*6250.032*	1825				
		37500		***Node Scheme:***			
1(bottom)	2	3	4-7	8	9	10 (top)	
0.2149	0.9126	1.1415	1.2837	0.9302	0.5842	0.1950	Profile
3.7303	**15.8433**	**19.8172**	**22.2858**[a]	**16.1487**	**10.1426**	**3.3862**	**Node Burns**

Source: Spreadsheet: abc, tab: time
Note a: Maximum burnup is 44,126(MWd/MTU) yielding a flux = 3.03E+13(neutrons/cm^2-sec)
Source: 5 wt. % U-235 case, File: P-37-5/NO4.output

Back checking Table 11 is done by multiplying each calculated burn value by the node length percentage and summing to regenerate the average power level as demonstrated in Table 12.

Table 12. Profile Burnup Verification

Nodes	**1 (bottom)**	**2**	**3**	**4-7**	**8**	**9**	**10 (top)**
Node Burns	3.7303	15.8433	19.8172	22.2858	16.1487	10.1426	3.3862
Node Fractions[a]	0.04	0.12	0.08	0.48	0.12	0.08	0.08
17.3612	0.1487	1.8949	1.6453	10.6620	1.9315	0.8087	0.2700

Source: Spreadsheet: abc, tab: time
Note a: More detail on Node Fractions is given in Appendix V-1

Column 4-7 in Table 12, represents the sum of 4 nodes, $0.12 \times 4 = 0.48$ fraction of the length of rod burned at a 22.2858 MW power level. The product: $0.47842 \times 22.2858 = 10.662$ is the average of the rod burnup for one cycle, times 6 cycles yields 17.3612 GWd. This procedure allows the analyst to confirm the selected rod average. The above procedure requires that the SAS2H runs be scaled to 1 metric ton (10^6 Kg.). The SAS2H models hold the pellet diameter, rod pitch and rod count to core dimensions; however, the rod length is adjusted to the desired mass (1 MTU). The rod lengths were calculated on Spreadsheet: abc, tab-Mass.

All of the SAS2H isotopic outputs are available electronically from the CD, Attachment VII. Conversions of SAS2H isotopic outputs to MCNP inputs use the same procedure as *Calculation of Isotopic Bias and Uncertainty for BWR SNF* (BSC 2003a, Section 5).

The MCNP model for Cooper (BSC 2003a) places the GE 7×7 SNF into a 44 BWR Waste Package in a flooded condition. The resulting k_{eff} data is presented in Table 13. A potential geometry overlap was identified during the checking process. The details of the potential overlap are given in Attachment V.2.

Input decks for all of the executed codes are available in Attachment VII.

System reactivity differences between the calculated and measured isotopic concentrations were determined with MCNP calculations. The MCNP calculations were performed to calculate the k_{eff} that results from using the RCA isotopic concentrations (k_{RCA}) in a flooded 44-BWR waste package. Only the isotopic concentrations were varied to the predicted concentrations from SAS2H (k_{SAS2H}) providing a comparison in terms of Δk_{eff} (k_{SAS2H}-k_{RCA}).

The 44-BWR waste package configuration follows the drawing provided in Attachment IV of BSC 2003a. The axial nodes from which the samples came from were used to represent fuel assemblies in a flooded waste package configuration. Axially reflective boundary conditions were used for each representation. The general assembly design parameters are presented in Table 13.

Table 13. Fuel Assembly Data Required for MCNP

Parameter	Cooper
Fuel Assembly Data	
Lattice	7x7
Number of Fuel Rods	49
Number of rods containing Gd_2O_3	5
Channel Tube Material	Zircaloy-4
Channel Tube Thickness (cm)	0.2
Assembly pitch (cm)	15.24
Fuel Rod Data	
Clad outer diameter (cm)	1.43
Clad thickness (cm)	0.188
Cladding material	Zircaloy-2
Rod pitch (cm)	1.875
Length (cm)*	370.84
Fuel Pellet Data	
Diameter (cm)	1.242
Pellet Material	UO_2
Pellet Initial Enrichment	2.939
Pellet Density	9.73

Source: Hermann and Dehart 1998, Cooper, pp.6, 8

The spent fuel isotopes used in the MCNP cases correspond to those from the SAS2H calculations and the measured sample isotopes. Isotopes were extracted from the SAS2H outputs and measured results, and then combined with the initial oxygen mass and renormalized to the total mass in terms of weight percents. The values from the SAS2H calculations are given in units of mols, which were converted to units of grams using Equation 7. The MCNP density input used the corresponding fresh fuel density. The SAS2H output files for each calculation are contained on a compact disc (Attachment VII).

$$Mass_i = (Mols\ Isotope_i) * A_i \quad \text{(Eq. 7)}$$

where ‘i’ is the particular isotope and A_i is the atomic mass value (Audi and Wapstra 1995).

The outer barrier of the waste package was represented as SB-575 N06022 as described in Table 14. The inner barrier was represented as SA-240 S31600, which is nuclear grade 316 stainless steel (SS) with tightened control on carbon and nitrogen content (ASM International 1987, p. 931; ASME 1998, Section II, SA-240, Table 1) as described in Table 15. The fuel basket plates were represented as Neutronit A978 with 1.62 weight percent boron as described in Table 16. See Attachment IV for use with Ni-Gd plates. Since the plates may eventually be made of a new Ni-Gd material, Alloy B932-04, Attachment IV was added to cover this possibility. The thermal shunts were represented as aluminum 6061 as described in Table 17. The basket side and corner guides were represented as Grade 70 A 516 carbon steel as described in Table 18. The basket stiffeners were represented as water since they are not solid over the length of the basket. The rod cladding was made of Zircaloy-2 as described in Table 19 and the channel tube material is Zircaloy-4 as described in Table 20.

The chromium, nickel, and iron elemental weight percents obtained from the references were expanded into their constituent natural isotopic weight percents for use in MCNP. This expansion was performed by: (1) calculating a natural weight fraction of each isotope in the elemental state, and (2) multiplying the elemental weight percent in the material of interest by the natural weight fraction of the isotope in the elemental state to obtain the weight percent of the isotope in the material of interest. This process is described mathematically in Equations 8 and 9. The atomic mass values and atom percent of natural element values for these calculations are from Parrington et al. (1996), and listed in files on the CD in directory Profile-Isos, Excel files P-12-3 through P-17-5.

$$\left(\begin{array}{c}\text{Weight Fraction}\\ \text{of Isotope "i" in the}\\ \text{Natural Element}\end{array}\right) = \frac{(\text{Atomic Mass of Isotope "i"})(\text{At\% of Isotope "i" in Natural Element})}{\sum_{i=1}^{I}(\text{Atomic Mass of Isotope "i"})(\text{At\% of Isotope "i" in Natural Element})} \quad \text{(Eq. 8)}$$

where 'I' is the total number of isotopes in the natural element

$$\left(\begin{array}{c}\text{Wt\% of Isotope "i" in}\\ \text{Material Composition}\end{array}\right) = \left(\begin{array}{c}\text{Weight Fraction}\\ \text{of Isotope "i" in the}\\ \text{Natural Element}\end{array}\right)\left(\begin{array}{c}\text{Reference Wt\% of}\\ \text{Element in Material Composition}\end{array}\right) \quad \text{(Eq. 9)}$$

Table 14. Material Specifications for SB-575 N06022 (Alloy 22)

Element/Isotope	ZAID	wt.%	Element/Isotope	ZAID	wt.%
C-nat	6000.50c	0.0150	^{59}Co	27059.50c	2.5000
^{55}Mn	25055.50c	0.5000	^{182}W	74182.55c	0.7877
Si-nat	14000.50c	0.0800	^{183}W	74183.55c	0.4278
^{50}Cr	24050.60c	0.8879	^{184}W	74184.55c	0.9209
^{52}Cr	24052.60c	17.7863	^{186}W	74186.55c	0.8636
^{53}Cr	24053.60c	2.0554	V	23000.50c	0.3500
^{54}Cr	24054.60c	0.5202	^{54}Fe	26054.60c	0.2260
^{58}Ni	28058.60c	36.8024	^{56}Fe	26056.60c	3.6759
^{60}Ni	28060.60c	14.6621	^{57}Fe	26057.60c	0.0865
^{61}Ni	28061.60c	0.6481	^{58}Fe	26058.60c	0.0116
^{62}Ni	28062.60c	2.0975	^{32}S	16032.50c	0.0200
^{64}Ni	28064.60c	0.5547	^{31}P	15031.50c	0.0200
Mo-nat	42000.50c	13.5000	Density = 8.69 g/cm^3		

Source: DTN: MO0003RIB00071.000

SA-240 S31600, is nuclear grade 316 stainless steel with tightened control on carbon and nitrogen content per ASM International 1987, p. 931; and ASME 1998, Section II, SA-240, Table 1).

Table 15. Material Specifications for SS316, SA-240 S31600

Element/Isotope	ZAID	wt.%	Element/Isotope	ZAID	wt.%
C-nat	6000.50c	0.0200	^{54}Fe	26054.60c	3.6911
^{14}N	7014.50c	0.0800	^{56}Fe	26056.60c	60.0322
Si-nat	14000.50c	1.0000	^{57}Fe	26057.60c	1.4119
^{31}P	15031.50c	0.0450	^{58}Fe	26058.60c	0.1897
^{32}S	16032.50c	0.0300	^{58}Ni	28058.60c	8.0641
^{50}Cr	24050.60c	0.7103	^{60}Ni	28060.60c	3.2127
^{52}Cr	24052.60c	14.2291	^{61}Ni	28061.60c	0.1420
^{53}Cr	24053.60c	1.6443	^{62}Ni	28062.60c	0.4596
^{54}Cr	24054.60c	0.4162	^{64}Ni	28064.60c	0.1216
^{55}Mn	25055.50c	2.0000	Mo-nat	42000.50c	2.5000
Density = 7.98 g/cm^3; Source: Density : ASTM G 1-90 1999, Table XI					

Source: ASM International (1987), p. 931, and ASME 1998, Section II, SA-240, Table 1.

Table 16. Material Specifications for Neutronit A978 with 1.62 weight percent Boron

Element/Isotope	ZAID	Wt%	Element/Isotope	ZAID	wt.%
^{10}B	5010.50c	0.2986	^{57}Fe	26057.60c	1.3928
^{11}B	5011.56c	1.3214	^{58}Fe	26058.60c	0.1872
C-nat	6000.50c	0.0400	^{59}Co	27059.50c	0.2000
^{50}Cr	24050.60c	0.7730	^{58}Ni	28058.60c	8.7361
^{52}Cr	24052.60c	15.4846	^{60}Ni	28060.60c	3.4805
^{53}Cr	24053.60c	1.7894	^{61}Ni	28061.60c	0.1539
^{54}Cr	24054.60c	0.4529	^{62}Ni	28062.60c	0.4979
^{54}Fe	26054.60c	3.6411	^{64}Ni	28064.60c	0.1317
^{56}Fe	26056.60c	59.2189	Mo-nat	42000.50c	2.2000
Density = 7.76 g/cm^3 ; Source: Kügler, A. 1996, Page 17					

Source of elemental composition: Kügler, A. 1991, Page 15
Source of Natural Isotopic percents: Parrington, 1996

Since the Neutronit A978 may not be the final poison material to use as plating, Attachment IV was added to demonstrate use of the 7 Node Profiles with a newer Alloy B932-04.

Table 17. Material Specifications for Al 6061

Element/Isotope	ZAID	wt.%	Element/Isotope	ZAID	wt.%
Si-nat	14000.50c	0.6000	Mg-nat	12000.50c	1.0000
^{54}Fe	26054.60c	0.0396	^{50}Cr	24050.60c	0.0081
^{56}Fe	26056.60c	0.6433	^{52}Cr	24052.60c	0.1632
^{57}Fe	26057.60c	0.0151	^{53}Cr	24053.60c	0.0189
^{58}Fe	26058.60c	0.0020	^{54}Cr	24054.60c	0.0048
^{63}Cu	29063.60c	0.1884	Ti-nat	22000.50c	0.1500
^{65}Cu	29065.60c	0.0866	^{27}Al	13027.50c	96.9300
^{55}Mn	25055.50c	0.1500	Density = 2.70 g/cm^3		

Source: Chemical Composition: ASM International. 1990, Table 2; Density : ASTM G 1-90 1999, Table XI

NOTE: Zn cross-section data unavailable, therefore it was substituted as ^{27}Al (See assumption 3.2).

Table 18. Material Specifications for Grade 70 A516 Carbon Steel

Element/Isotope	ZAID	wt.%[a]	Element/Isotope	ZAID	wt.%[a]
C-nat	6000.50c	0.2700	^{54}Fe	26054.60c	5.5558
^{55}Mn	25055.50c	1.0450	^{56}Fe	26056.60c	90.3584
^{31}P	15031.50c	0.0350	^{57}Fe	26057.60c	2.1252
^{32}S	16032.50c	0.0350	^{58}Fe	26058.60c	0.2856
Si-nat	14000.50c	0.2900	Density[b] = 7.850 g/cm^3		

Sources: [a] ASTM A 516/A 516M-01, 2001,Table 1
[b] ASTM A 20/A20M-99a, 1999, p. 9

Table 19. Material Specifications for Zircaloy-2

Element/Isotope	ZAID	wt. %	Element/Isotope	ZAID	wt.%
^{16}O	6000.50c	0.1250	^{58}Fe	26058.60c	0.0004
^{50}Cr	24050.60c	0.0042	^{58}Ni	28058.60c	0.0370
^{52}Cr	24052.60c	0.0837	^{60}Ni	28060.60c	0.0147
^{53}Cr	24053.60c	0.0097	^{61}Ni	28061.60c	0.0007
^{54}Cr	24054.60c	0.0024	^{62}Ni	28062.60c	0.0021
^{54}Fe	26054.60c	0.0076	^{64}Ni	28064.60c	0.0006
^{56}Fe	26056.60c	0.1241	Sn-nat	50000.35c	1.4500
^{57}Fe	26057.60c	0.0029	Zr-nat	40000.60c	98.1350
Density = 6.55 g/cm^3 Source: ASM International 1967. *Alloy Digest,* (July), Section 2.021					

Source: ASTM B 811-90. 1991, Table 2

Table 20. Material Specifications for Zircaloy-4

Element/Isotope	ZAID	wt.%	Element/Isotope	ZAID	wt.%
^{50}Cr	24050.60c	0.0042	^{57}Fe	26057.60c	0.0045
^{52}Cr	24052.60c	0.0837	^{58}Fe	26058.60c	0.0006
^{53}Cr	24053.60c	0.0097	^{16}O	8016.50c	0.1250
^{54}Cr	24054.60c	0.0024	Zr-nat	40000.60c	98.1150
^{54}Fe	26054.60c	0.0119	Sn-nat	50000.35c	1.4500
^{56}Fe	26056.60c	0.1930	Density = 6.56 g/cm^3		

Source: Chemical Composition: ASTM B 811-90. 1991,Table 2; Density: ASM International. 1990, page 666,Table 6

MCNP Input Conversions

The SAS2H codes project the concentrations of the isotopic inventories and the MCNP code scales the inputs to the selected density. However, an important component of the SNF is the oxygen concentration. Oxygen is initially present in the form of the uranium oxide UO_2. Oxygen does not significant react with neutrons and it is assumed to remain constant through out the irradiation process (Assumption 3.2). Using this property of oxygen, the remaining isotope concentrations can be scaled from the SAS2H output to appropriate MCNP input values. MCNP will scale the values to the specified density value: 9.37 g/cm^3-Cooper (BSC 2003a, Section 5.2.1), 9.213 g/cm^3 LaSalle (CRWMS M&O 1999b, Table 4-302) as shown in Table 33.

Equation 4 was derived to project the oxygen concentration in the SNF based on the initial oxygen concentration.

$$C(O) = \frac{wt.\%O_i * \Sigma_i C_i}{1 - wt.\%O_i}$$

(Eq. 10)

where $C(O)$ = Projected concentration of oxygen
wt. % O_i = Initial oxygen Weight Percent in UO_2 pellet
$\Sigma_i C_i$ = Summation of all SAS2H projected concentrations of isotopes of interest

With a projection of the oxygen concentration and the sum of the concentrations of each isotope of interest from SAS2H, the normalized MCNP inputs are computed with Equation 5.

$$\text{MCNP Input}_i = 100 * \left(\frac{C_i}{C_t} \right) \quad \text{(Eq. 11)}$$

where MCNP Input_i = The MCNP input value for each individual isotope
C_i = SAS2H projected concentration of the individual isotope
C_t = The sum of all the SAS2H projected concentrations plus the projected concentration of oxygen.

MCNP Demonstration Results

Table 21 provides the results of the Cooper BWR 7x7 demonstration.

Table 21. MCNP Results for a Flooded BWR WP

File	k_{eff}	Sigma[a]	AENCF[b] (ev)
P-12-3o	0.72598	0.00059	0.231
P-12-4o	0.78424	0.00073	0.214
P-12-5o	0.83154	0.0007	0.203
P-17-3o	0.69801	0.00056	0.239
P-17-4o	0.76835	0.00066	0.218
P-17-5o	0.81705	0.00067	0.206
P-22-3o	0.68364	0.00063	0.241
P-22-4o	0.7446	0.00068	0.224
P-22-5o	0.79705	0.00063	0.211
P-27-3o	0.66678	0.00056	0.247
P-27-4o	0.72425	0.00065	0.228
P-27-5o	0.77052	0.00067	0.216
P-32-3o	0.65083	0.00055	0.253
P-32-4o	0.69565	0.00064	0.236
P-32-5o	0.75101	0.00069	0.223
P-37-3o	0.64051	0.00054	0.255
P-37-4o	0.69987	0.00064	0.234
P-37-5o	0.74384	0.00060	0.222

Source: Spreadsheet: keff, tab: 12-10-03
Note a: Sigma is the reported error associated with the statistically based Monte Carlo calculation.
Note b: AENCF is the Average energy of a neutron causing fission in electron volts (ev), traditionally reported with results.

Figure 4 illustrates the results of the demonstration. As expected, the multiplication of the waste packages drops with increasing burnup and increases with the initial enrichment of the fuel.

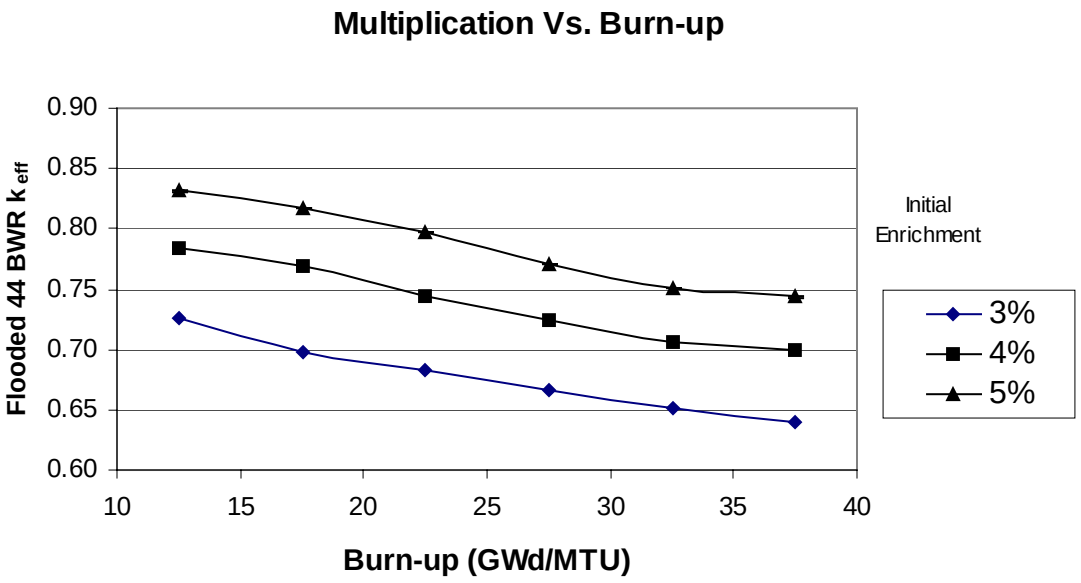

Figure 4. MCNP Results of Demonstration Runs (Flooded 44 BWR WP)

Source: Spreadsheet: keff, tab: sheet 1

5.4 MINIMUM MODERATOR DENSITY

It was of interest to quantify the minimum average moderator density over the fuel length. Since this is a direct application of the spreadsheets developed for this project, it is included here. The procedure was straightforward. The CRC data was previously broken into groupings by burnup. Each of the six-burnup groups identified the average and minimum moderator density, corresponding to the $T_{95/95}$ value of the group per Table A-31 in Experimental Statistics (Natrella 1963). Table 22 presents the minimum moderator densities of the six groups.

Table 22. Minimum Moderator Densities

Burnup Ranges	Node 1 (top)	Node 2	Node 3	Node 4	Node 5	Node 6	Node 7	Node 8	Node 9	Node 10 (bottom)
10-14.9	0.6928	0.6613	0.4582	0.3191	0.2412	0.1899	0.1497	0.1236	0.1280	0.1095
15-19.9	0.6948	0.6625	0.4371	0.2975	0.2199	0.1897	0.1545	0.1280	0.1333	0.1140
20-24.9	0.6976	0.6691	0.4602	0.3194	0.2391	0.2038	0.1621	0.1347	0.1410	0.1207
25-29.9	0.7022	0.6796	0.4816	0.3375	0.2538	0.2183	0.1603	0.1395	0.1565	0.1351
30-34.9	0.7005	0.6748	0.4730	0.3325	0.2513	0.2172	0.1734	0.1438	0.1493	0.1284
35-40	0.7082	0.6881	0.5584	0.4295	0.3282	0.2565	0.2033	0.1716	0.1721	0.1531

Source: Spreadsheet: GGTH-Node, tab: Mod-sort

The above nodes were averaged together, multiplied by the node lengths to determine the weights of each node. The weights were summed together for the weighted average of the minimum moderator density or the $T_{95/95}$ minimum as shown in Table 23. Therefore, the $T_{95/95}$ value for the average minimum moderator density is 0.30.

Table 23. Weighted Average Minimum Density $T_{95/95}$

Burnup Ranges	Node 1 (bottom)	Node 2	Node 3	Node 4	Node 5	Node 6	Node 7	Node 8	Node 9	Node 10 (top)	
Average:	0.6993	0.6726	0.4781	0.3393	0.2556	0.2126	0.1672	0.1402	0.1467	0.1268	Weighted
Node:	0.04	0.12	0.08	0.12	0.12	0.12	0.12	0.12	0.08	0.08	Average
Weights:	0.0280	0.0807	0.0398	0.0407	0.0307	0.0255	0.0201	0.0168	0.0117	0.0101	0.3042

Source: Spreadsheet: GGTH-Node, tab: Mod-sort

5.5 OTHER TEMPERATURES

To utilize this calculation a criticality safety analyst will select the $T_{95/95}$ burnup values from Table 4, the average $T_{95/95}$ moderator density from Table 7 and average fuel temperature from Table 9. The analyst would still require the clad, moderator and gap temperatures to complete SAS2H inputs for calculations of the depleted isotopic concentrations. These temperatures are not provided in the CRC data. These additional temperatures do not have a significant impact on the depletion codes; however, they have been added for completeness.

Grand Gulf (Table 2-1, Punatar 2001) presented both the core inlet (Ci) temperature (278.9 C) and the core saturation (Cs) temperature (287.7 C). The bottom node was left at the core inlet temperature and it was assumed that the core would reach saturation near the 8th node (Assumption 3.3). This leaves 68% of the rod length to estimate a temperature gradient with a linear fit. The Grand Gulf case is shown as an example in Table 24 with the 8.8 C temperature difference (about 13 degrees F) linearly spread over the nodes.

Table 24. Moderator Temperature

Node	% Length	Moderator Temperature	Moderator Temperature	Grand Gulf Example (K)
10 (top)	8%	Cs	Cs	560.9
9	8%	Cs	Cs	560.9
8	12%	Cs	Cs	560.9
4-7	48%	=(42/68)*(Cs-Ci) + Ci	=38.2%Ci + 61.8% Cs	557.5
3	8%	=(16/68)*(Cs-Ci) + Ci	=76.5%Ci + 23.5% Cs	554.1
2	12%	=(6/68)*(Cs-Ci) + Ci	=91.2%Ci + 8.8% Cs	552.9
1 (bottom)	4%	Ci	Ci	552.1

Source: Spreadsheet: temp, tab: Sheet1

Since the core inlet and outlet temperatures are set reactor parameters, an analyst can determine the moderator temperatures for all reactor nodes.

A methodology for calculating the cladding temperature is presented in Section 5.3.2 of BSC 2003b. Obviously, the cladding temperature would be less than the provided fuel temperature and higher than the moderator temperature ($T_f > T_c > T_m$). The analysts knows the power level (P in kilowatts, required for SAS2H inputs) and the moderator temperature (T_m). The equations presented in Section 5.3.2 (BSC 2003b) and Equation 8-119 from Todreas and Kazimi (1990) are simplified to a set of equations presented as Equation 12.

$$\begin{aligned} T_f - T_m &= (A' + B' + C') * P \\ T_f - T_g &= A' * P \\ T_g - T_c &= B' * P \\ T_c - T_m &= C' * P \end{aligned} \qquad \text{(Eq. 12)}$$

with: T_f = Average Fuel Temperature (K);
T_m = Average Moderator Temperature (K);
T_g = Average Gap Temperature (K);
T_c = Average Clad Temperature (K);
P = Power (kW);
A', B' and C' = Arbitrary Constants

Inspections of calculated values (figure 8-17, Todreas and Kazimi 1990) of the associated temperatures drops across the fuel pellet, gap, cladding and moderator suggest approximately a 60, 30, 10% split. Matching figure 8-17 (Todreas and Kazimi 1990) can be done by redefining the arbitrary constants from Eq. 12 to: A'=0.6A, B'=0.3A and C'=0.1A. Substituting the new constants into Equation 12 and solving for A gives Equation 13:

$$A = \frac{T_f - T_m}{P} \qquad \text{(Eq. 13)}$$

Notice that when solving for the cladding and gap temperatures, the power levels divide out.

$$\begin{aligned} T_c &= 0.1T_f + 0.9T_m \\ T_g &= 0.4T_f + 0.6T_m \end{aligned} \qquad \text{(Eq. 14)}$$

Using an example, Grand Gulf Assembly A10, statepoint 10 (Punatar, 2001), with a burnup of 28.39 GWd/MTU and assuming four cycles at 360 days would imply a power level of 19.72 MW/MTU. Applying the Bounding BWR Profile from Table 4 to the example generates the power per node, shown in Table 25. Table 25 also provides the moderator density from Table 7, the fuel temperature from Table 9 and the moderator temperature from Table 24. Table 25 also calculates the clad and gap temperature from Equation 14.

Table 25. Example Data Set

Node	BWR Profile	Power per Node (MW/MTU)	Mod. Density (g/cc)	Fuel Temp (K)	Gap Temp (K)	Clad Temp (K)	Mod. Temp (K)
10 (top)	0.1976	3.90	0.1972	623	523	567	561
9	0.5514	10.87	0.2058	744	560	579	561
8	0.8480	16.72	0.2176	835	587	588	561
4-7	1.3017	25.67	0.3314	898	604	592	557
3	1.1611	22.90	0.5711	877	596	586	554
2	0.9334	18.41	0.7199	831	581	581	553
1(bottom)	0.2016	3.98	0.7432	629	520	560	552

Source: Spreadsheet temp; tab: Sheet 2

Figure 5 illustrates the temperature profiles determined for the above example.

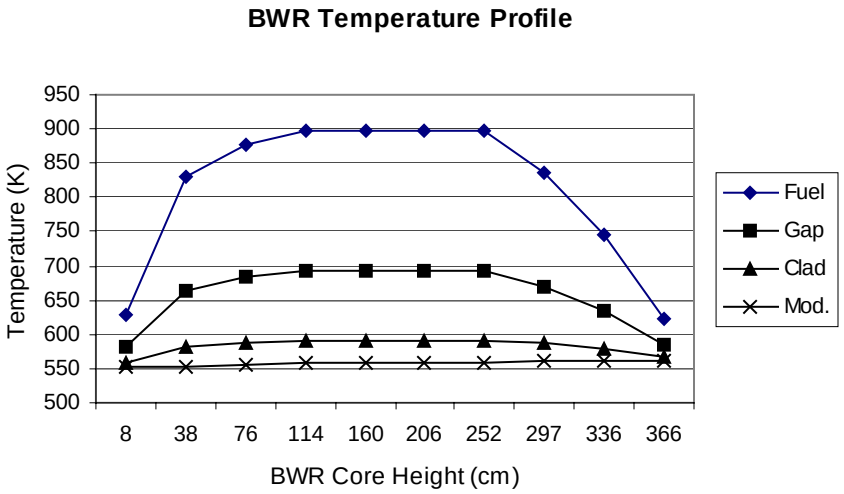

Figure 5. BWR Temperature Profile

Source: Spreadsheet Temp; tab: Sheet 2

5.6 LASALLE REACTOR TEST

The calculations in Section 5.3 show the developed profiles behave as expected. To test that the profiles are conservative to actual SNF, detailed analysis of reactor conditions are required. The LaSalle reactor was chosen as the test case. Assemblies from each of the six burnup ranges were selected. The 25-node data available from *Summary Report of Commercial Reactor Data for LaSalle Unit 1* (CRWMS M&O, 1999b) was used. Tables 26-31 presents this 25-node data (control data) for each assembly selected adjacent to the 7-node layout and a direct 7-node compression of the data. The last section of each table presents the profile data that is being tested to verify that the profiles produce bounding eigenvalues of the waste package effective neutron multiplication factors (k_{effs}).

A formal hypothesis, test procedure and passing criteria for the test are presented in Attachment III.

Table 26. Selected LaSalle Statepoint representing 10-14.9 GWd/MTU Range

LaSalle Assembly A1 Datapoint 3 (BOC Cy5)					Length	381	cm	Average Burnup		12.425	(GWd/MTU)			
25-Node Statepoint					7-Node Layout			7-Node Statepoint			7-Node Profile			
25-Node Num.	Node Length	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	7-Node Num.	%	Length (cm)	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	Burnup Profile	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)
1	15.24	2.348	648.5	0.7396	1	4%	15.24	2.348	648.5	0.7396	0.1782	2.214	638	0.7425
2	15.24	8.028	910.7	0.7396										
3	15.24	11.389	1028.6	0.7300	2	12%	45.72	11.025	1030.2	0.7191	0.9112	11.322	950	0.7060
4	15.24	13.657	1151.3	0.6878										
5	15.24	14.846	1198.5	0.6317	3	8%	30.48	15.138	1201.1	0.6017	1.1901	14.787	1023	0.5214
6	15.24	15.429	1203.7	0.5716										
7	15.24	15.728	1195.8	0.5150										
8	15.24	15.928	1188.0	0.4656										
9	15.24	16.474	1201.2	0.4234										
10	15.24	16.436	1194.2	0.3871										
11	15.24	16.276	1188.2	0.3565	4-7	48%	182.88	15.481	1170.6	0.3400	1.3266	16.482	1007	0.2858
12	15.24	16.062	1184.5	0.3303										
13	15.24	15.853	1184.4	0.3079										
14	15.24	15.614	1183.4	0.2885										
15	15.24	15.284	1175.3	0.2716										
16	15.24	14.806	1155.1	0.2569										
17	15.24	14.167	1123.2	0.2441										
18	15.24	13.147	1073.6	0.2329										
19	15.24	12.425	1034.4	0.2234										
20	15.24	11.727	996.5	0.2152	8	12%	45.72	11.680	994.7	0.2155	0.7978	9.913	908	0.1822
21	15.24	10.888	953.2	0.2079										
22	15.24	9.625	894.3	0.2017	9	8%	30.48	8.913	862.9	0.1992	0.5096	6.331	789	0.1732
23	15.24	8.201	831.5	0.1967										
24	15.24	3.882	680.4	0.1930	10	8%	30.48	3.147	656.8	0.1923	0.1883	2.340	631	0.1635
25 (top)	15.24	2.411	633.2	0.1915										

Source: CRWMS M&O 1999b, Section 4 Spreadsheet: LaSalle-Runs, tab: 10-14.9

Table 27. Selected LaSalle Statepoint representing 15-19.9 GWd/MTU Range

LaSalle Assembly A6 Datapoint 4 (239.5 Cy5)					Length	381	cm	Average Burnup		17.739 (GWd/MTU)				
25-Node Statepoint					7-Node Layout			7-Node Statepoint			7-Node Profile			
25-Node Num.	Node Length	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	7-Node Num.	%	Length (cm)	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	Burnup Profile	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)
1	15.24	3.547	629.3	0.7396	1	4%	15.24	3.547	629.3	0.7396	0.1880	3.335	640	0.7463
2	15.24	12.572	847.0	0.7396										
3	15.24	17.836	915.5	0.7352	2	12%	45.72	17.117	908.4	0.7247	0.9405	16.683	940	0.6984
4	15.24	20.944	962.7	0.6994										
5	15.24	22.341	974.9	0.6499	3	8%	30.48	22.606	976.2	0.6224	1.1901	21.111	997	0.5114
6	15.24	22.871	977.4	0.5949										
7	15.24	23.041	978.3	0.5417										
8	15.24	23.129	979.9	0.4943										
9	15.24	23.710	990.9	0.4531										
10	15.24	23.556	991.4	0.4177										
11	15.24	23.243	990.0	0.3876	4-7	48%	182.88	21.934	970.6	0.3709	1.3154	23.335	972	0.2780
12	15.24	22.830	986.9	0.3620										
13	15.24	22.345	982.1	0.3398										
14	15.24	21.793	975.2	0.3206										
15	15.24	21.173	965.9	0.3038										
16	15.24	20.466	953.7	0.2890										
17	15.24	19.602	937.3	0.2761										
18	15.24	18.317	915.0	0.2649										
19	15.24	17.300	888.2	0.2548										
20	15.24	16.254	857.0	0.2459	8	12%	45.72	16.157	854.3	0.2462	0.8112	14.390	865	0.1790
21	15.24	14.918	817.7	0.2380										
22	15.24	12.983	772.2	0.2312	9	8%	30.48	11.877	748.1	0.2284	0.5136	9.112	756	0.1689
23	15.24	10.771	723.9	0.2256										
24	15.24	4.961	630.5	0.2216	10	8%	30.48	3.971	615.8	0.2206	0.1822	3.232	623	0.1598
25 (top)	15.24	2.981	601.0	0.2195										

Source: CRWMS M&O 1999b, Section 4 Spreadsheet: LaSalle-Runs, tab: 15-19.9

Table 28. Selected LaSalle Statepoint representing 20-24.9 GWd/MTU Range

LaSalle Assembly A2 Datapoint 5 (BOC Cy5)					Length	381	cm	Average Burnup	22.938	(GWd/MTU)				
25-Node Statepoint					7-Node Layout			7-Node Statepoint			7-Node Profile			
25-Node	Node	Burnup	Fuel	Mod. Dens.	7-Node	%	Length	Burnup	Fuel	Mod. Dens.	Burnup	Burnup	Fuel	Mod. Dens.
Num.	Length	(GWd/MTU)	Temp. (K)	(g/cm^3)	Num.		(cm)	(GWd/MTU)	Temp. (K)	(g/cm^3)	Profile	(GWd/MTU)	Temp. (K)	(g/cm^3)
1	15.24	4.573	614.5	0.7396	1	4%	15.24	4.573	614.5	0.7396	0.1950	4.473	632	0.7467
2	15.24	16.061	772.0	0.7396										
3	15.24	21.967	795.3	0.7385	2	12%	45.72	21.098	795.8	0.7288	0.9443	21.659	870	0.7026
4	15.24	25.265	820.1	0.7082										
5	15.24	26.822	832.7	0.6657	3	8%	30.48	27.196	837.6	0.6414	1.1869	27.224	926	0.5302
6	15.24	27.570	842.5	0.6171										
7	15.24	27.998	852.5	0.5685										
8	15.24	28.352	863.5	0.5239										
9	15.24	29.278	881.6	0.4840										
10	15.24	29.372	892.9	0.4494										
11	15.24	29.299	903.3	0.4193	4-7	48%	182.88	28.170	911.7	0.4004	1.3089	30.023	920	0.2928
12	15.24	29.102	913.0	0.3933										
13	15.24	28.815	922.4	0.3706										
14	15.24	28.440	931.3	0.3506										
15	15.24	27.971	938.9	0.3329										
16	15.24	27.388	944.7	0.3173										
17	15.24	26.622	947.9	0.3033										
18	15.24	25.404	948.1	0.2914										
19	15.24	24.286	939.6	0.2802										
20	15.24	22.953	921.1	0.2700	8	12%	45.72	22.755	914.8	0.2703	0.8201	18.812	832	0.1872
21	15.24	21.025	883.6	0.2606										
22	15.24	18.432	840.4	0.2529	9	8%	30.48	16.875	812.1	0.2496	0.5176	11.871	739	0.1767
23	15.24	15.318	783.7	0.2462										
24	15.24	7.001	653.0	0.2412	10	8%	30.48	5.564	632.8	0.2396	0.1982	4.546	620	0.1689
25 (top)	15.24	4.127	612.5	0.2380										

Source: CRWMS M&O 1999b, Section 4 Spreadsheet: LaSalle-Runs, tab: 20-24.9

Table 29. Selected LaSalle Statepoint representing 25-29.9 GWd/MTU Range

LaSalle Assembly A5 Datapoint 5 (BOC Cy5)					Length	381	cm	Average Burnup 27.476			(GWd/MTU)			
25-Node Statepoint					7-Node Layout			7-Node Statepoint			7-Node Profile			
25-Node Num.	Node Length	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	7-Node Num.	%	Length (cm)	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	Burnup Profile	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)
1	15.24	5.842	659.0	0.7396	1	4%	15.24	5.842	659.0	0.7396	0.2017	5.542	629	0.7496
2	15.24	20.269	956.4	0.7396										
3	15.24	27.241	995.8	0.7267	2	12%	45.72	26.155	994.6	0.7162	0.9339	25.661	831	0.7048
4	15.24	30.954	1031.5	0.6824										
5	15.24	32.572	1040.4	0.6259	3	8%	30.48	32.909	1042.4	0.5965	1.1618	31.921	877	0.5375
6	15.24	33.245	1044.3	0.5671										
7	15.24	33.597	1050.3	0.5132										
8	15.24	33.924	1059.4	0.4667										
9	15.24	34.897	1079.4	0.4269										
10	15.24	34.946	1089.8	0.3925										
11	15.24	34.781	1100.3	0.3630	4-7	48%	182.88	33.450	1110.4	0.3461	1.3024	35.785	898	0.2997
12	15.24	34.473	1111.0	0.3379										
13	15.24	34.067	1121.6	0.3161										
14	15.24	33.583	1131.5	0.2971										
15	15.24	33.022	1140.0	0.2804										
16	15.24	32.359	1146.3	0.2657										
17	15.24	31.522	1149.0	0.2525										
18	15.24	30.232	1146.4	0.2408										
19	15.24	28.990	1128.1	0.2306										
20	15.24	27.455	1092.3	0.2214	8	12%	45.72	27.185	1082.3	0.2219	0.8484	23.311	835	0.1919
21	15.24	25.109	1026.4	0.2136										
22	15.24	21.991	954.3	0.2069	9	8%	30.48	20.124	912.3	0.2043	0.5517	15.158	744	0.1808
23	15.24	18.257	870.2	0.2016										
24	15.24	8.487	689.7	0.1979	10	8%	30.48	6.788	662.3	0.1971	0.1977	5.431	623	0.1733
25 (top)	15.24	5.088	634.8	0.1962										

Source: CRWMS M&O 1999b, Section 4 Spreadsheet: LaSalle-Runs, tab: 25-29.9

Table 30. Selected LaSalle Statepoint representing 30-34.9 GWd/MTU Range

LaSalle Assembly A7 Datapoint 6 (196.1 Cy6)					Length	381	cm	Average Burnup 32.351	(GWd/MTU)					
25-Node Statepoint					7-Node Layout			7-Node Statepoint			7-Node Profile			
25-Node Num.	Node Length	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	7-Node Num.	%	Length (cm)	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	Burnup Profile	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)
1	15.24	6.920	614.0	0.7396	1	4%	15.24	6.920	614.0	0.7396	0.2042	6.606	613	0.7388
2	15.24	23.739	744.9	0.7396										
3	15.24	31.477	757.9	0.7274	2	12%	45.72	30.259	761.4	0.7171	0.9174	29.680	738	0.7167
4	15.24	35.561	781.5	0.6843										
5	15.24	37.467	804.1	0.6299	3	8%	30.48	37.947	815.7	0.6022	1.1328	36.646	766	0.5851
6	15.24	38.426	827.3	0.5745										
7	15.24	39.042	848.2	0.5240										
8	15.24	39.573	864.8	0.4800										
9	15.24	40.793	883.6	0.4415										
10	15.24	40.968	892.9	0.4077										
11	15.24	40.877	897.0	0.3783	4-7	48%	182.88	39.388	877.2	0.3595	1.2896	41.721	792	0.3413
12	15.24	40.606	896.5	0.3527										
13	15.24	40.225	893.1	0.3303										
14	15.24	39.743	887.4	0.3107										
15	15.24	39.139	880.3	0.2934										
16	15.24	38.374	871.4	0.2782										
17	15.24	37.380	860.9	0.2647										
18	15.24	35.936	850.0	0.2529										
19	15.24	34.534	837.7	0.2423										
20	15.24	32.823	823.5	0.2330	8	12%	45.72	32.493	820.1	0.2333	0.8918	28.851	762	0.2175
21	15.24	30.121	799.2	0.2247										
22	15.24	26.541	770.1	0.2179	9	8%	30.48	24.315	749.5	0.2149	0.6013	19.453	701	0.2014
23	15.24	22.088	728.9	0.2119										
24	15.24	10.299	634.8	0.2078	10	8%	30.48	8.212	618.3	0.2062	0.2122	6.866	610	0.1905
25 (top)	15.24	6.125	601.7	0.2046										

Source: CRWMS M&O 1999b, Section 4 Spreadsheet: LaSalle-Runs, tab: 30-34.9

Table 31. Selected LaSalle Statepoint representing 35-40 GWd/MTU Range

LaSalle Assembly A1 Datapoint10 (196.1 Cy6)					Length	381	cm	Average Burnup	39.741	(GWd/MTU)				
25-Node Statepoint					7-Node Layout			7-Node Statepoint			7-Node Profile			
25-Node Num.	Node Length	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	7-Node Num.	%	Length (cm)	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)	Burnup Profile	Burnup (GWd/MTU)	Fuel Temp. (K)	Mod. Dens. (g/cm^3)
1	15.24	8.841	580.3	0.7396	1	4%	15.24	8.841	580.3	0.7396	0.2149	8.540	615	0.7294
2	15.24	29.450	616.9	0.7396										
3	15.24	38.075	617.5	0.7347	2	12%	45.72	36.627	618.0	0.7253	0.9126	36.268	732	0.7146
4	15.24	42.356	619.7	0.7015										
5	15.24	44.332	622.8	0.6585	3	8%	30.48	44.857	625.1	0.6358	1.1415	45.364	758	0.594
6	15.24	45.381	627.4	0.6130										
7	15.24	46.144	633.9	0.5700										
8	15.24	46.922	641.5	0.5317										
9	15.24	48.576	651.4	0.4977										
10	15.24	49.066	661.0	0.4681										
11	15.24	49.276	671.5	0.4424	4-7	48%	182.88	48.144	691.7	0.4248	1.2837	51.016	810	0.3598
12	15.24	49.317	682.9	0.4198										
13	15.24	49.267	695.0	0.3998										
14	15.24	49.101	707.4	0.3819										
15	15.24	48.769	720.2	0.3660										
16	15.24	48.211	732.9	0.3518										
17	15.24	47.289	745.3	0.3393										
18	15.24	45.792	757.4	0.3287										
19	15.24	44.168	765.5	0.3182										
20	15.24	42.083	768.1	0.3083	8	12%	45.72	41.640	764.3	0.3084	0.9302	36.967	793	0.2412
21	15.24	38.668	759.4	0.2988										
22	15.24	34.188	743.8	0.2912	9	8%	30.48	31.183	726.7	0.2870	0.5842	23.217	705	0.2273
23	15.24	28.177	709.5	0.2828										
24	15.24	12.789	624.0	0.2753	10	8%	30.48	10.039	608.7	0.2710	0.1950	7.750	616	0.2171
25 (top)	15.24	7.289	593.3	0.2666										

Source: CRWMS M&O 1999b, Section 4 Spreadsheet: LaSalle-Runs, tab: 35-40

To develop SAS2H inputs of the LaSalle assemblies the following data tables were assembled.

U-235 assays were provided in CRWMS M&O 1999b, Table 3-3, equations for estimating the U-234 and U-236 content of fresh fuel were used from Equation 15 (Bowman et al. 1995 p.20).

$$U^{234}wt\% = (0.007731) * (U^{235}wt\%)^{1.0837}$$

$$U^{236}wt\% = (0.0046) * (U^{235}wt\%)$$

$$U^{238}wt\% = 100 - U^{234}wt\% - U^{235}wt\% - U^{236}wt\%$$

(Eq. 15)

Table 32 provides the uranium isotopic information calculated from Eq. 15.

Table 32. LaSalle Type 9 "A" Fuel Assay

Fuel # / U Isotope	wt. % U-235	wt. % U-234	wt. % U-236	wt. % U-238
100 & 101- Nat	0.71	0.005	0.003	99.280
102	3.50	0.030	0.016	96.454
103 & 104	3.65	0.031	0.017	96.302
105	3.50	0.030	0.016	96.454

Source of U-235 enrichment data: CRWMS M&O 1999b, Table 3-3
Spreadsheet: LaSalle-Runs, tab: Assay

The density of the LaSalle type 9 is provided in CRWMS M&O 1999b, Table 4-302, as Rod Linear Mass. CRWMS M&O 1999b, Table 2-2 provides fuel diameters. The fuel density is found by dividing the Rod Linear Mass by πr^2, as shown in Table 33.

Table 33. LaSalle Type 9 "A" Fuel Data

LaSalle Fuel		UO_2		Rod Linear Mass	
	Diameter (cm)	Radius (cm)		g/cm/rod	Density(g/cc)
Pellet	1.04394	0.52197		7.88567	9.213
Inner Clad	1.06426	0.53213			
Outer Clad	1.2268	0.6134			

Source: CRWMS M&O 1999b, Table 2-2 & 4-302
Spreadsheet: LaSalle-Runs, tab: Index

Additionally, BWRs use varying assays and gadolinium oxide (Gd_2O_3, also referred to as gadolinia) enrichments. Averaging the gadolinia from gad pins over the fuel in the assembly provides the wt % Gadolinia. A spreadsheet 'gad-cal' was developed specifically to average the gadolinium and generate the required SAS2H input card. Table 34 tabulates the fuel types, enrichments and gadolinia weight percents.

Table 34. Gadolinia Weight Percent

		Uranium		Gadolinia	Gadolinia
Fuel Type	**# of Rods**	**Enrichment**	**# of Gd Rods**	**Enrichment**	**wt. %**
100	48	0.71	0	0	0
101	60	0.71	0	0	0
102	60	3.50	4/6	5.0/4.0	0.73
103	60	3.65	6/6	5.0/4.0	0.87
104	60	3.65	4/6	5.0/4.0	0.73
105	60	3.50	7/3	5.0/4.0	0.78

Source: CRWMS M&O 1999b, Table 3-3; Spreadsheet: LaSalle-runs, tab: Index

The fuel types are used in various axial layouts as presented in Table 35.

Table 35. 25-Node Axial Fuel Type Layout

	Type	Uranium	Gadolinia	Number
Nodes	**9 (A)**	**Enrichment**	**Enrichment**	**of Fuel Rods**
25 (top)	100	0.71	0	48
24	101	0.71	0	60
23	102	3.50	0.73	60
22	102	3.50	0.73	60
21	102	3.50	0.73	60
20	103	3.65	0.87	60
19	103	3.65	0.87	60
18	103	3.65	0.87	60
17	104	3.65	0.73	60
16	104	3.65	0.73	60
15	104	3.65	0.73	60
14	104	3.65	0.73	60
13	104	3.65	0.73	60
12	104	3.65	0.73	60
11	104	3.65	0.73	60
10	104	3.65	0.73	60
9	104	3.65	0.73	60
8	105	3.50	0.78	60
7	105	3.50	0.78	60
6	105	3.50	0.78	60
5	105	3.50	0.78	60
4	105	3.50	0.78	60
3	105	3.50	0.78	60
2	105	3.50	0.78	60
1 (Bottom)	101	0.71	0	60

Source: CRWMS M&O 1999b, Tables 3-3 & 3-4.
Spreadsheet: gad-calc, tab: Gd-cal

For both the 7-node CRC compression (7N) and the 7-node profile (7P) version of the CRC data, the following fuel chart in Table 36 was used.

Table 36. 7-Node Axial Layout

	Type	Uranium	Gadolinia	Number
Nodes	**9 (A)**	**Enrichment**	**Enrichment**	**of Fuel Rods**
10 (Top)	100	0.71	0	48
9	102	3.50	0.73	60
8	103	3.65	0.87	60
4-7	104	3.65	0.73	60
3	105	3.50	0.78	60
2	105	3.50	0.78	60
1 (Bottom)	101	0.71	0	60

Source: CRWMS M&O 1999b, Tables 3-3 & 3-4.
Spreadsheet: LaSalle-Runs, tab: Index

Additional data required for the development of the SAS2H input deck is provided in Table 37.

Note that the simplified version uses 3.65 wt.% uranium enrichment for all assays and 60 rods, with no gadolinia.

Table 37. LaSalle Unit 1 Fuel Assembly Core Data

Description	Data
Fuel assembly array size and types	GE 8x8NB (GE9B)
Number of water rods	1
Number of assemblies in core	764
Number of control blades in core	185
Active Fuel Height	381 cm
Pin Pitch	1.6256 cm
Assembly Pitch	15.24 cm
Water Rod outside diameter	3.4036
Water Rod inside diameter	3.2004
Water Rod/Channel Material	Zircaloy
Channel - inner width thickness	0.254 cm
Reference moderator Density	0.7396 g/cm^3

Source: CRWMS M&O 1999b, Table 2-2
Spreadsheet: LaSalle-Runs, tab: Index

These SAS2H input decks, when executed, will produce the isotopic inventory of the modeled SNF. For the LaSalle test case, the ORIGEN-S code was executed multiple times for each SAS2H run. This process simulated aging the SNF for 0, 5, 10, 33, 100, 333, 1,000, 3,333, 5,000, 10,000, 11,000, 12,000, 13,000, 14,000, 15,000, 16,000, 18,000, 20,000, 22,500, 25,000, 27,500, 30,000, 35,000, 50,000 75,000, and 100,000 years. However; only data from the 5-year isotopic inventories were used in the MCNP calculations, all of the isotopic inventories for the above time steps were retained with the records for the LaSalle calculations.

The calendar burn days were obtained from Table 3-6 of CRWMS M&O 1999b for each assembly chosen. Additionally, the average power levels for each burn period were slightly different; therefore, the individual node burns were scaled such that the overall total matched the CRC data and they were proportional to the average power levels reported.

Table 38. LaSalle Unit 1 Operating Cycle History for Cycles 4-8

			Calendar		
Cycle	**Beginning**	**End**	**Days**	**Shutdown**	**EFPD[a]**
4	1/5/1990	2/16/1991	407		375.32
5	5/12/1991	10/3/1992	510	85	467.14
6	1/30/1993	2/18/1994	384	119	316.01
7	6/24/1994	1/25/1996	580	126	495.23
8	4/21/1996			87	

Source: CRWMS M&O 1999b, Table 3-6
Note a: EFPD is Effective Full Power Days

To judge the effects of the parameters used in SAS2H input, a parametric study was performed. Documented in Attachment II, the parametric study shows that both burnup and moderator densities were the primary factors on the final WP k_{eff}. The study used the actual calendar burn days of the six burnup groups with moderator coolant densities in steps: 0.1, 0.2, 0.3, 0.4, 0.5, 0.6 and 0.7.

Once the LaSalle SAS2H inputs were developed and executed, a MCNP input deck of a 44-BWR waste package was used to determine the k_{eff}. A version was used which incorporated a detailed neutron source term, starting neutrons in specific locations throughout the WP.

While the LaSalle assemblies have a significant amount of information available for analysis, the criticality safety analyst will probably not have that much information. It is anticipated that the analyst would be provided a maximum assay and average burnup of the assembly. The details of assay and gadolinia enrichments would not be available, additionally time and computing restraints would not allow full-detailed computations in a production environment. Therefore, a second set of runs were executed with all uranium enrichments of 3.65 wt.% U-235 and all gadolinia removed from the mixture. The set of runs with all of the detailed information was denoted “Detailed”, and the set of runs that simulated the anticipated level of data were denoted “Simplified”. Table 39 provides the results of the detailed cases, the 25-node CRC data (control group), the 7-node compression (7N) of the CRC data and the 7-node profile (7P) test cases.

The raw data from the test runs is presented in Table 39.

Table 39. MCNP k_{eff} Results of the LaSalle Test

File[b]	k_{eff}	sigma	AENCF[a] (ev)	File	k_{eff}	sigma	AENCF[a] (ev)
25-Node CRC CONTROL				7-Node CRC Compressed			
25-12-Do	0.70817	0.00048	0.212	7N-12-Do	0.70325	0.00046	0.217
25-12-No	0.69689	0.00049	0.205	7N-12-No	0.70418	0.0005	0.209
25-18-Do	0.68096	0.00042	0.221	7N-18-Do	0.65548	0.00042	0.240
25-18-No	0.68051	0.00048	0.210	7N-18-No	0.66568	0.00041	0.221
25-23-Do	0.64546	0.00045	0.237	7N-23-Do	0.64453	0.00045	0.244
25-23-No	0.7031	0.00045	0.213	7N-23-No	0.66105	0.00048	0.224
25-27-Do	0.64808	0.00043	0.240	7N-27-Do	0.64882	0.00044	0.248
25-27-No	0.64331	0.00041	0.221	7N-27-No	0.64773	0.00044	0.228
25-32-Do	0.62013	0.00046	0.252	7N-32-Do	0.62637	0.00042	0.258
25-32-No	0.62712	0.00042	0.226	7N-32-No	0.63184	0.00047	0.235
25-39-Do	0.55529	0.00037	0.284	7N-39-Do	0.55794	0.00039	0.290
25-39-No	0.60257	0.00044	0.234	7N-39-No	0.60577	0.00048	0.242
7-Node Profile Test							
7P-12-Do	0.71552	0.00046	0.214				
7P-12-No	0.7171	0.00046	0.205				
7P-18-Do	0.68811	0.00048	0.230				
7P-18-No	0.68319	0.00044	0.216				
7P-23-Do	0.68494	0.00045	0.230				
7P-23-No	0.71936	0.00049	0.204				
7P-27-Do	0.66981	0.00045	0.238				
7P-27-No	0.66878	0.00048	0.221				
7P-32-Do	0.64575	0.00048	0.248				
7P-32-No	0.64968	0.00043	0.228				
7P-39-Do	0.62659	0.00045	0.259				
7P-39-No	0.63557	0.00046	0.232				

Source: Spreadsheet; LaSalle-Runs, tab: keff
Note a: AENCF stands for Average Energy of a Neutron Causing Fission in electron-volts (ev)
Note b: Files with “D” are the detailed cases; with “N” are the simplified cases –N for “No-Gad”

Table 40 presents the data for the detailed cases, the 25-node CRC data (control group), the 7-node compression (7N) of the CRC data and the 7-node profile (7P) test cases.

Table 40. MCNP k_{eff} Results of Detailed Cases

Representing Burnup Group[a]	Actual Burnup Value[a]	CONTROL 25-Detailed	Compressed 7N-Detailed	Percent Error	TEST 7P-Detailed	Percent Higher
10-14.9	12.425	0.70817	0.70325	0.69%	0.71552	1.04%
15-19.9	17.739	0.68096	0.65548	3.74%	0.68811	1.05%
20-24.9	22.938	0.64546	0.64453	0.14%	0.68494	6.12%
25-29.9	27.476	0.64808	0.64882	-0.11%	0.66981	3.35%
30-34.9	32.35	0.62013	0.62637	-1.01%	0.64575	4.13%
35-40	39.74	0.55529	0.55794	-0.48%	0.62659	12.84%

Note a: Units of the burnup are in (GWd/MTU).
Source: Spreadsheet; LaSalle-Runs, tab: keff

Figure 6 demonstrates the results of the detailed cases.

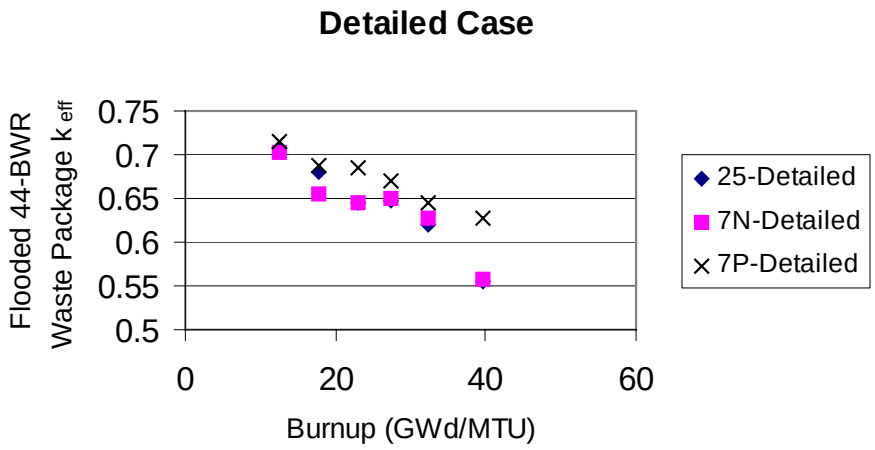

Figure 6. MCNP results of Detailed Case

Source: Spreadsheet; LaSalle-Runs, tab: keff

Note that the burnups are of different assemblies that were modeled in detail with significantly different burnup histories to arrive at the final isotopic composition. Therefore, it is not expected that the points fall into smooth trend lines as in the previous demonstration. The charts are presented to illustrate the profile case (7P) are conservative to both the 25-node CRC data and the 7-node compression of the 25-node CRC data.

Table 41 provides the results of the simplified cases, all with an assay of 3.65 wt.% and no gadolinia present during initial loading. Presented are the 25-node CRC data (control group), the 7-node compression (7N) of the CRC data and the 7-node profile (7P) test cases.

Table 41. MCNP k_{eff} Results of Simplified Cases

Representing Burnup Group[a]	Actual Burnup Value[a]	CONTROL 25-Simplified	Compressed 7N- Simplified	Percent Error	TEST 7P-Simplified	Percent Higher
10-14.9	12.425	0.69689	0.70418	-1.05%	0.7171	2.90%
15-19.9	17.739	0.68051	0.66568	2.18%	0.68319	0.39%
20-24.9	22.938	0.7031	0.66105	5.98%	0.71936	2.31%
25-29.9	27.476	0.64331	0.64773	-0.69%	0.66878	3.96%
30-34.9	32.35	0.62712	0.63184	-0.75%	0.64968	3.60%
35-40	39.74	0.60257	0.60577	-0.53%	0.63557	5.48%

Note a: Units of the burnup are in (GWd/MTU).
Source: Spreadsheet; LaSalle-Runs, tab: keff

Figure 7 demonstrates the results of the simplified cases.

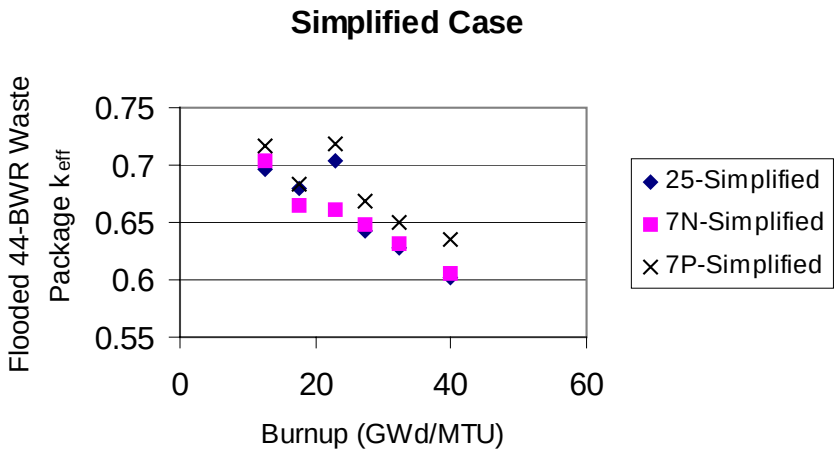

Figure 7. MCNP results of Simplified Case

Source: Spreadsheet; LaSalle-Runs, tab: keff

5.7 LASALLE REACTOR TEST RESULTS

The passing criteria established in Attachment III required all the profile cases to be above the 25-node CRC control cases. The 'percent higher' column in Tables 40 and 41 clearly demonstrate that all of the profile cases are more reactive (conservative) than the control cases. Therefore, the test cases pass the stated hypothesis in Attachment III:

"The statistically developed burnup and moderator density profiles derived from the available CRC data bases can be used to conservatively predict the effective neutron multiplication factor (k_{eff}) of SNF in flooded 44-BWR waste packages."

Furthermore, the 7-node profile cases are very close to the 25-node CRC cases in the low burnup range, and tend to drift to higher (more conservative) k_{eff} values with higher burnup. The results indicate very good, but conservative, agreement in low burnups ranges where k_{eff} values are of concern. In higher burnup ranges with lower k_{eff} values, the profiles provide more conservative results.

Per the passing criteria in Attachment III:

1. The error between the 25-node CRC and 7-node CRC compressed data should average less than 1%.

   The average error in the detailed 7-node CRC detailed compressed data is 0.50% and the simplified compressed data is 0.86%

2. The 7-node profiles are conservative to both the 25-node CRC and the 7-node CRC compressed, for all data points as presented in Tables 40 and 41.

3. The simplified data was expected to be more reactive, and it was extremely close to the detailed cases averaging only about 2% more reactive.

4. The test does justify the statistically derived profiles as a viable approach to analytically bound BWR SNF.

5. The 7-node compressed cases trend very well with the 25-node CRC controls, in most cases with very close results. One case out of the 18, the simplified 23 GWd/MTU case, showed both the 7-node compression and the 7-node profile case to be 9% more reactive.

6. Attachment III gives the technical version of the safety principle simply stated as: *"Safety shall not be based on only one calculation"*. This test processed six assemblies, each from the available CRC 25-Node data each with a unique burn-up history. The test procedure then simulated loading a 44-BWR assembly completely with each of the six SNF assemblies. The accident condition of complete flooding was assumed. Figure 6 demonstrates that the developed 7 Node Profiles proved conservative k_{effs} in six out of six of these cases.

   Additionally, Attachment IV repeated the 12 MCNP cases (six 25-Node CRC control and six 7-Node Profile) changing only the poison plate material. Attachment IV further demonstrates that use of the 7-Node Profiles are conservative to the 25 Node CRC data.

6. RESULTS

This analysis develops bounding profiles representing BWR axial burnups from a statistical analysis of extensive data on three different BWR cores. Table 4 presents the bounding profiles that will allow a criticality safety analyst to conservatively assign burnup values to axial sections of BWR SNF. The corresponding values for the moderator density are found in Table 7 and the fuel temperature is found in Table 10.

The methodology for determining moderator temperatures is provided in Table 24. The methodology for determining gap and clad temperatures is provided by Equation 12. Additionally, the average minimum moderator density for an entire assembly is presented in Table 23.

Compiling the presented information, the following six worksheets, Tables 42-47, are presented to assist the user. For a particular assembly, the individual power cycles must be determined from the burnup and cycle time. Given that the power (MW/MTU) is determined for the assembly, the user may multiply by the appropriate profile value to determine the power for each node. Double checking the value is suggested and is done by multiplying the node power by the percentage length of the node and summing the values. In all cases, this should return the selected power for the assembly. The only other information that is required for SAS2H input development is the coolant inlet temperature (Ci) and the coolant saturation temperature (Cs) which are usually published with the reactor information. A linear fit is provided between the coolant inlet and saturation temperatures. The Gap and Clad temperatures can then be estimated from the provided linear equations.

Tables 42 through 47 are compilations of Tables 4,7, 9 and 24. The tables also utilize Equation 14.

Table 42. 10-14.9 GWd/MTU Burnup Worksheet

10-14.9 GWd/MTU		Power per	Check -	$T_{95/95}$ Mod.	Fuel	Gap	Clad	Mod.
Node	% Length	Node (MW/MTU)	X	Density (g/cc)	Temp. (K)	Temp (K)	Temp (K)	Temp (K)
10	8%	0.1883 * *P*		0.1854	631	$0.4T_f+0.6T_m$	$0.1T_f+0.9T_m$	Cs
9	8%	0.5096 * *P*		0.1955	789	" "	" "	Cs
8	12%	0.7978 * *P*		0.2059	908	" "	" "	Cs
4-7	48%	1.3266 * *P*		0.3169	1007	" "	" "	38.2%Ci + 61.8% Cs
3	8%	1.1901 * *P*		0.5607	1023	" "	" "	76.5%Ci + 23.5% Cs
2	12%	0.9112 * *P*		0.7090	950	" "	" "	91.2%Ci + 8.8% Cs
1	4%	0.1782 * *P*		0.7311	638	" "	" "	Ci

Note: Sum of Power per Node times the percent of length columns (labeled 'X') should return the initial Power (*P*).

Table 43. 15-19.9 GWd/MTU Burnup Worksheet

15-19.9 GWd/MTU		Power per	Check -	$T_{95/95}$ Mod.	Fuel	Gap	Clad	Mod.
Node	% Length	Node (MW/MTU)	X	Density (g/cc)	Temp. (K)	Temp (K)	Temp (K)	Temp (K)
10	8%	0.1822 * *P*		0.1785	623	$0.4T_f+0.6T_m$	$0.1T_f+0.9T_m$	Cs
9	8%	0.5136 * *P*		0.1883	756	" "	" "	Cs
8	12%	0.8112 * *P*		0.1982	865	" "	" "	Cs
4-7	48%	1.3154 * *P*		0.3042	972	" "	" "	38.2%Ci + 61.8% Cs
3	8%	1.1901 * *P*		0.5449	997	" "	" "	76.5%Ci + 23.5% Cs
2	12%	0.9405 * *P*		0.7102	940	" "	" "	91.2%Ci + 8.8% Cs
1	4%	0.1880 * *P*		0.7388	640	" "	" "	Ci

Note: Sum of Power per Node times the percent of length columns (labeled 'X') should return the initial Power (*P*).

Table 44. 20-24.9 GWd/MTU Burnup Worksheet

20-24.9 GWd/MTU		Power per	Check -	$T_{95/95}$ Mod.	Fuel	Gap	Clad	Mod.
Node	% Length	Node (MW/MTU)	X	Density (g/cc)	Temp. (K)	Temp (K)	Temp (K)	Temp (K)
10	8%	0.1982 * *P*		0.1937	620	$0.4T_f+0.6T_m$	$0.1T_f+0.9T_m$	Cs
9	8%	0.5176 * *P*		0.2026	739	" "	" "	Cs
8	12%	0.8201 * *P*		0.2137	832	" "	" "	Cs
4-7	48%	1.3089 * *P*		0.3272	920	" "	" "	38.2%Ci + 61.8% Cs
3	8%	1.1869 * *P*		0.5667	926	" "	" "	76.5%Ci + 23.5% Cs
2	12%	0.9443 * *P*		0.7146	870	" "	" "	91.2%Ci + 8.8% Cs
1	4%	0.1950 * *P*		0.7381	632	" "	" "	Ci

Note: Sum of Power per Node times the percent of length columns (labeled 'X') should return the initial Power (*P*).

Table 45. 25-29.9 GWd/MTU Burnup Worksheet

25-29.9 GWd/MTU		Power per	Check -	$T_{95/95}$ Mod.	Fuel	Gap	Clad	Mod.
Node	% Length	Node (MW/MTU)	X	Density (g/cc)	Temp. (K)	Temp (K)	Temp (K)	Temp (K)
10	8%	0.1977 * *P*		0.1972	623	$0.4T_f+0.6T_m$	$0.1T_f+0.9T_m$	Cs
9	8%	0.5517 * *P*		0.2058	744	" "	" "	Cs
8	12%	0.8484 * *P*		0.2176	835	" "	" "	Cs
4-7	48%	1.3024 * *P*		0.3314	898	" "	" "	38.2%Ci + 61.8% Cs
3	8%	1.1618 * *P*		0.5711	877	" "	" "	76.5%Ci + 23.5% Cs
2	12%	0.9339 * *P*		0.7199	831	" "	" "	91.2%Ci + 8.8% Cs
1	4%	0.2017 * *P*		0.7432	629	" "	" "	Ci

Note: Sum of Power per Node times the percent of length columns (labeled 'X') should return the initial Power (*P*).

Table 46. 30-34.9 GWd/MTU Burnup Worksheet

30-34.9 GWd/MTU		Power per	Check -	$T_{95/95}$ Mod.	Fuel	Gap	Clad	Mod.
Node	% Length	Node (MW/MTU)	X	Density (g/cc)	Temp. (K)	Temp (K)	Temp (K)	Temp (K)
10	8%	0.2122 * *P*		0.2011	610	$0.4T_f+0.6T_m$	$0.1T_f+0.9T_m$	Cs
9	8%	0.6013 * *P*		0.2121	701	" "	" "	Cs
8	12%	0.8918 * *P*		0.2279	762	" "	" "	Cs
4-7	48%	1.2896 * *P*		0.3542	792	" "	" "	38.2%Ci + 61.8% Cs
3	8%	1.1328 * *P*		0.5945	766	" "	" "	76.5%Ci + 23.5% Cs
2	12%	0.9174 * *P*		0.7077	738	" "	" "	91.2%Ci + 8.8% Cs
1	4%	0.2042 * *P*		0.7221	613	" "	" "	Ci

Note: Sum of Power per Node times the percent of length columns (labeled 'X') should return the initial Power (*P*).

Table 47. 35-40 GWd/MTU Burnup Worksheet

35-40 GWd/MTU		Power per	Check -	$T_{95/95}$ Mod.	Fuel	Gap	Clad	Mod.
Node	% Length	Node (MW/MTU)	X	Density (g/cc)	Temp. (K)	Temp (K)	Temp (K)	Temp (K)
10	8%	0.1950 * *P*		0.2171	616	$0.4T_f+0.6T_m$	$0.1T_f+0.9T_m$	Cs
9	8%	0.5842 * *P*		0.2273	705	" "	" "	Cs
8	12%	0.9302 * *P*		0.2412	793	" "	" "	Cs
4-7	48%	1.2837 * *P*		0.3598	810	" "	" "	38.2%Ci + 61.8% Cs
3	8%	1.1415 * *P*		0.5940	758	" "	" "	76.5%Ci + 23.5% Cs
2	12%	0.9126 * *P*		0.7146	732	" "	" "	91.2%Ci + 8.8% Cs
1	4%	0.2149 * *P*		0.7294	615	" "	" "	Ci

Note: Sum of Power per Node times the percent of length columns (labeled 'X') should return the initial Power (*P*).

The outputs are reasonable compared to the inputs and the results are suitable for the intended use. The SAS2H and MCNP input and output files used in this evaluation are contained on an attached CD to this calculation file, Attachment VII, as listed in Attachment VI.

Uncertainties with this statistical approach are inherent. The $T_{95/95}$ limit implies that 5% of the projected node burnups are underestimated. Equation 16 is derived from tree probability statistics.

$$Odds = Coeff.* base_1^{\exp_1} * base_2^{\exp_2} \quad \text{(Eq. 16)}$$

with:
Odds = Odds of an individual outcome
Coeff. = Total number of combinations that reach each outcome
base = The odds of the individual event; $base_1$ = 95% and $base_2$ = 5% for the $T_{95/95}$ Limit
exp = Exponential, the number of events with that outcome.

A single parameter statistical study of the $T_{95/95}$ limit is developed in Table 48 from Equation 16.

Table 48. Odds Table for Single-Parameter 7-Node $T_{95/95}$ Statistics

Coeff.	$base_1$	exp_1	$base_2$	exp_2	Odds	Percentage	Notes
1	0.95	7	0.05	0	0.698	69.8%	All 7-nodes were within $T_{95/95}$
7	0.95	6	0.05	1	0.257	25.7%	1 of the 7-nodes was outside $T_{95/95}$
21	0.95	5	0.05	2	0.041	4.1%	2 of the 7-nodes were outside $T_{95/95}$
35	0.95	4	0.05	3	3.56E-03	0.4%	3 of the 7-nodes were outside $T_{95/95}$
35	0.95	3	0.05	4	1.88E-04	0.0%	4 of the 7-nodes were outside $T_{95/95}$
21	0.95	2	0.05	5	5.92E-06	0.0%	5 of the 7-nodes were outside $T_{95/95}$
7	0.95	1	0.05	6	1.04E-07	0.0%	6 of the 7-nodes were outside $T_{95/95}$
1	0.95	0	0.05	7	7.81E-10	0.0%	All 7-nodes were outside $T_{95/95}$
			Total Odds		1	100.0%	

Source: GGTH-node.xls, tab: Odds

This study showed that two parameters have a strong influence on waste package k_{eff}. Both the burnup and the BWR moderator/coolant density were taken at the $T_{95/95}$ bounding levels. The above single parameter odds were used to develop a dual parameter odds table.

The case "All 7-nodes were within $T_{95/95}$" has odds of $0.95^7 = 70\%$. This case is repeated to find the odds of all 14-nodes within $T_{95/95}$. There is only one way all 14 parameters could be within the $T_{95/95}$ bounds, with odds of $0.95^7 * 0.95^7 = 49\%$. Equation 17 demonstrates this use of tree probability.

$$Combined\_Odds = Coeff.* odds_1 * odds_2$$
$$TotalOdds = \sum_n Combined\_Odds$$

(Eq. 17)

with:
Combined_Odds = Odds of a specific combination for an individual outcome.
Coeff. = Number of combinations that reach each outcome
$odds_n$ = Odds of each individual outcome, from Table 48.
Total Odds = Sum of the different combinations that give the same outcomes

For example: In one case 7 burnups are within $T_{95/95}$ bounds, and 5 moderator densities are within $T_{95/95}$ bounds and vice versa; giving 2 cases (Coeff.=2) of 12. Additionally, 6 burnups and 6 moderator densities are within $T_{95/95}$ bounds, (Coeff. = 1). Making the total odds of 12 of the 14 parameters within the $T_{95/95}$ bounds equal to: (2*69.8%*4.1%) + (1*25.7%*25.7%) = 5.72% + 6.62% = 12.33%.

Table 49 compiles all the possible combinations of 7-node dual parameters, both at the $T_{95/95}$ bounding limit.

Table 49. Probability Tree for Dual-Parameter 7-Node $T_{95/95}$ Statistics

Combination	Coeff.	Number	$odds_1$	$odds_2$	Comb. Odds
7-7	1	14	6.983E-01	6.983E-01	4.877E-01
7-6	2	13	6.983E-01	2.573E-01	3.593E-01
7-5	2	12	6.983E-01	4.062E-02	5.674E-02
7-4	2	11	6.983E-01	3.563E-03	4.977E-03
7-3	2	10	6.983E-01	1.876E-04	2.619E-04
7-2	2	9	6.983E-01	5.923E-06	8.272E-06
7-1	2	8	6.983E-01	1.039E-07	1.451E-07
7-0	2	7	6.983E-01	7.813E-10	1.091E-09
6-6	1	12	2.573E-01	2.573E-01	6.619E-02
6-5	2	11	2.573E-01	4.062E-02	2.090E-02
6-4	2	10	2.573E-01	3.563E-03	1.834E-03
6-3	2	9	2.573E-01	1.876E-04	9.651E-05
6-2	2	8	2.573E-01	5.923E-06	3.048E-06
6-1	2	7	2.573E-01	1.039E-07	5.347E-08
6-0	2	6	2.573E-01	7.813E-10	4.020E-10
5-5	1	10	4.062E-02	4.062E-02	1.650E-03
5-4	2	9	4.062E-02	3.563E-03	2.895E-04
5-3	2	8	4.062E-02	1.876E-04	1.524E-05
5-2	2	7	4.062E-02	5.923E-06	4.812E-07
5-1	2	6	4.062E-02	1.039E-07	8.442E-09
5-0	2	5	4.062E-02	7.813E-10	6.347E-11
4-4	1	8	3.563E-03	3.563E-03	1.270E-05
4-3	2	7	3.563E-03	1.876E-04	1.337E-06
4-2	2	6	3.563E-03	5.923E-06	4.221E-08
4-1	2	5	3.563E-03	1.039E-07	7.405E-10
4-0	2	4	3.563E-03	7.813E-10	5.568E-12
3-3	1	6	1.876E-04	1.876E-04	3.518E-08
3-2	2	5	1.876E-04	5.923E-06	2.222E-09
3-1	2	4	1.876E-04	1.039E-07	3.898E-11
3-0	2	3	1.876E-04	7.813E-10	2.930E-13
2-2	1	4	5.923E-06	5.923E-06	3.508E-11
2-1	2	3	5.923E-06	1.039E-07	1.231E-12
2-0	2	2	5.923E-06	7.813E-10	9.254E-15
1-1	1	2	1.039E-07	1.039E-07	1.080E-14
1-0	2	1	1.039E-07	7.813E-10	1.624E-16
0-0	1	0	7.813E-10	7.813E-10	6.104E-19
				Total Odds:	1

Source: GGTh-Node.xls, tab: Odds

Table 50 compiles the data from the tree presented in Table 49, by completing the summation for total odds prescribed in Equation 17.

Table 50. Odds Table for Dual-Parameter 7-Node $T_{95/95}$ Statistics

Number	Total Odds	Notes
14	48.8%	All 14 node parameters (NPs) were within $T_{95/95}$
13	35.9%	1 of the 14 NPs was outside the $T_{95/95}$
12	12.3%	2 of the 14 NPs were outside the $T_{95/95}$
11	2.6%	3 of the 14 NPs were outside the $T_{95/95}$
10	0.4%	4 of the 14 NPs were outside the $T_{95/95}$
9	0.0%	5 of the 14 NPs were outside the $T_{95/95}$
8	0.0%	6 of the 14 NPs were outside the $T_{95/95}$
7	0.0%	7 of the 14 NPs were outside the $T_{95/95}$
6	0.0%	8 of the 14 NPs were outside the $T_{95/95}$
5	0.0%	9 of the 14 NPs were outside the $T_{95/95}$
4	0.0%	10 of the 14 NPs were outside the $T_{95/95}$
3	0.0%	11 of the 14 NPs were outside the $T_{95/95}$
2	0.0%	12 of the 14 NPs were outside the $T_{95/95}$
1	0.0%	13 of the 14 NPs were outside the $T_{95/95}$
0	0.0%	All 14 NPs were outside $T_{95/95}$
Total	100.0%	

Source: GGTH-node.xls, tab: Odds

Tables 49 and 50 present odds of two 7-node parameters. The center burnup node labeled “Nodes 4-7” was not determined by the $T_{95/95}$ statistics. Six of the seven burnup nodes were taken at the $T_{95/95}$ minimum burnup limit with all seven moderator densities taken at the $T_{95/95}$ minimum density, slightly improving the odds presented in Table 50.

Therefore, Table 50 supports the conclusion that half of assemblies analyzed with these profiles will have all parameters bound by the burnup and moderator density profiles. The odds of having more than two of the 14 parameters outside the $T_{95/95}$ bounding limit are vanishing small.

To determine the odds on the final WP k_{eff} being bound by the resulting calculations utilizing the profile data would require a statically significant number of tests for each burnup group to be completed. The analysis presented in this calculation demonstrates that the k_{eff} of a flooded 44 BWR WP loaded with SNF is bound by the profiles for six out of six assemblies.

7. REFERENCES

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7.2 CODES, STANDARDS, PROCEDURES AND REGULATIONS

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7.3 DATA USED BY DATA TRACKING NUMBER

MO0003RIB00071.000. Physical and Chemical Characteristics of Alloy 22. Submittal date: 03/13/2000.

7.4 SOFTWARE CODES

CRWMS M&O 1998c. *Software Code: MCNP.* V4B2LV. HP, HPUX 9.07 and 10.20; PC, Windows 95; Sun, Solaris 2.6. 30033 V4B2LV.

CRWMS M&O 2000c. *Software Code: SCALE.* V4.4A. HP. 10129-4.4A-00.

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ATTACHMENT I

COSINE PROFILE

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ATTACHMENT I: COSINE PROFILE

The term Axial Profile (or Burnup Profile) is used to denote a mathematical representation of the fuel utilization or "burnup" inside a nuclear reactor. With no reflection, the maximum use of the nuclear fuel is at the center of the core, with utilization falling off rapidly to zero at the top and bottom of the core. This is referred to as the Cosine Profile, similar to a cosine function from trigonometry with the peak at pi radians (180^o) and zero at the origin and at twice the value of pi (360^o).

With an average burnup value and the appropriate axial profile, a user will be able to generate the specific burnup values along the axial length of a reactor core. The cosine example is presented in Figure I-1.

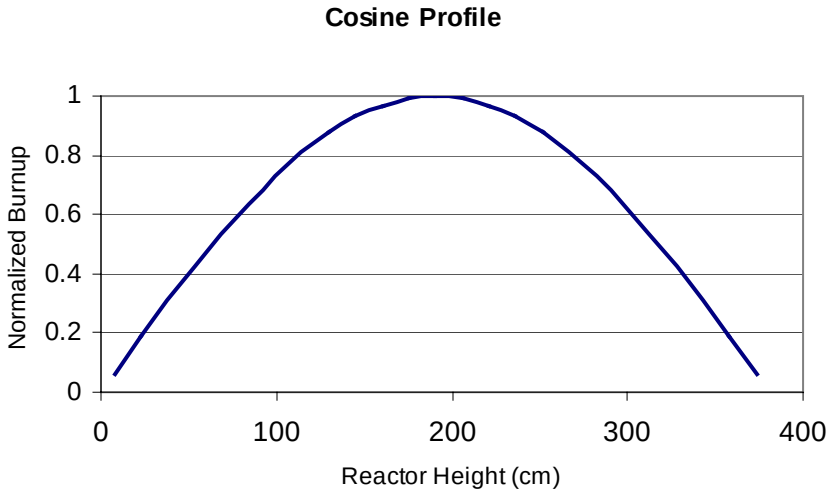

Figure I-1. Idealized Cosine Profile

Source: Spreadsheet Cosine, tab: Cosine

The above individual node values can be found from the equation for the above curve:

$$b = B\cos(\frac{\pi}{2} - \frac{\pi h}{L})$$

Eq. I-1

with : b = individual burnup value (GWd/MTU) (*dependent variable*)
B = Maximum Burnup of the assembly (GWd/MTU)
h = Height of the node (cm) (*independent variable*)
L = Total height of the assembly (cm)

However, the user may be provided an average burnup value ($\bar{b}$) and not the maximum. From the equation, a relationship can be derived.

Setting the area under the curve equal to a rectangle representing the average burnup:

$$\int_0^L B\cos(\frac{\pi}{2} - \frac{\pi h}{L})dh = \bar{b}L$$

Eq.I-2

Completing the integration and solving for B:

$$B = \frac{\pi}{2}\bar{b}$$

Eq. I-3

Then for a provided average burnup value for an assembly and knowing the profile we can quickly find the maximum burnup from Equation I-3 and the individual burnup for each node from Equation I-1. For example, if a 31.83 GWd/MTU burnup were reported with the assembly, a maximum of 50 GWd/MTU would have burned at the center. Figure I-2 demonstrates this relationship and illustrates that the area under the profile is equal to the area under the average line.

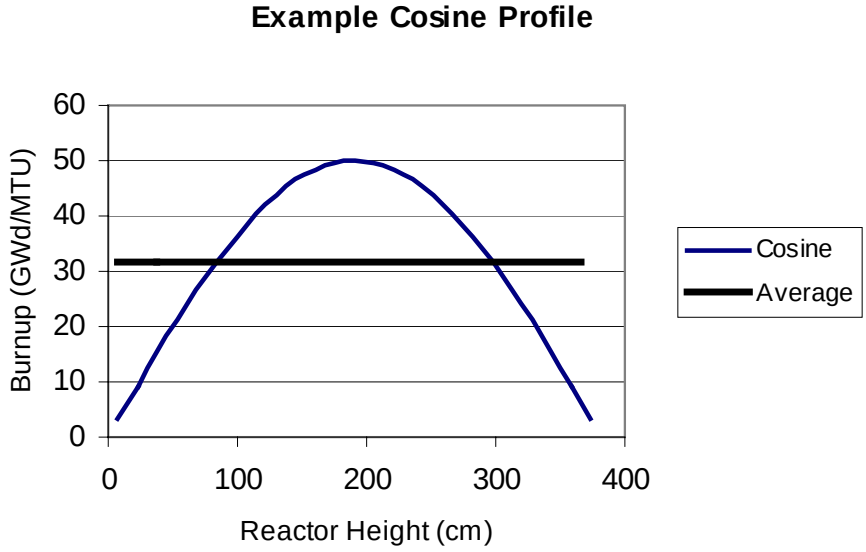

Figure I-2. Example Cosine Profile

Source: Spreadsheet Cosine, tab: Cosine

It is of interest to demonstrate the node schemes used to mathematically depict nuclear reactor cores utilizing the idealized cosine profile. The CRC data, presented in CRWMS M&O 1999a; CRWMS M&O 1999b; and Punatar 2001 provided by the utilities is usually in a 25-node layout, as shown in Figure I-3.

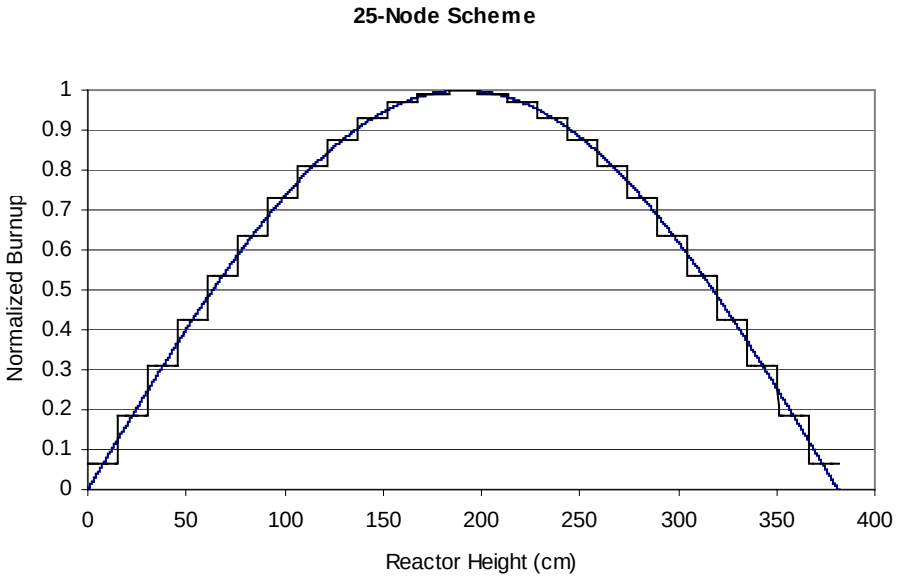

Figure I-3. 25-Node Scheme superimposed on an idealized cosine burnup

Source: Spreadsheet Cosine, tab: Graph

The three references compress the data into a 10-node format as shown in Figure I-4.

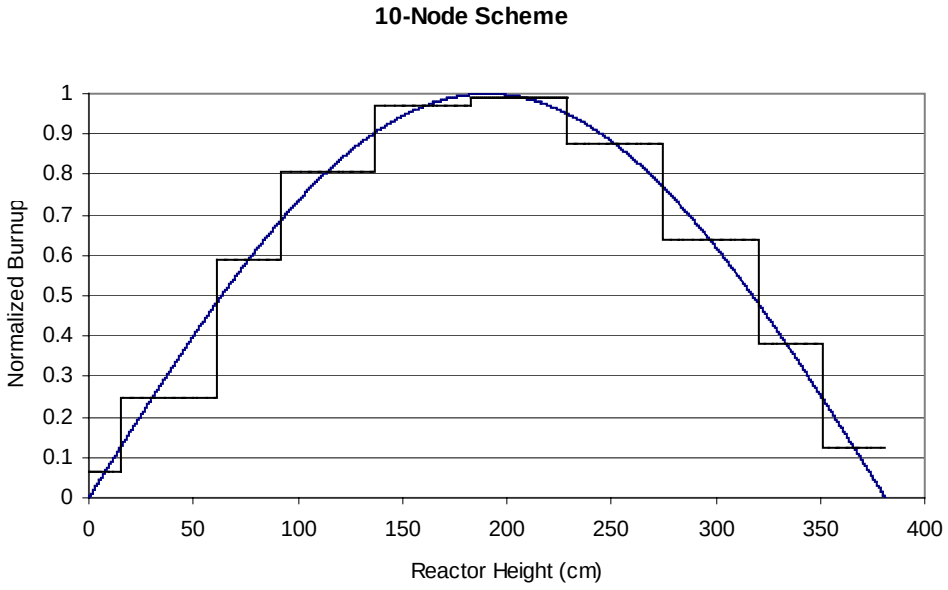

Figure I-4. 10-Node scheme superimposed on an idealized cosine burnup

Source: Spreadsheet Cosine, tab: Graph

This report further condenses the 10-node schemes into a 7-node scheme by taking $T_{95/95}$ values for six outer nodes and compensating the four inner nodes to conserve overall burnup. However, it is necessary to demonstrate that a 7-node compression is mathematically sound. Figure I-5,

demonstrates a 7-node scheme which simply averages the four inner nodes of the 10-node scheme.

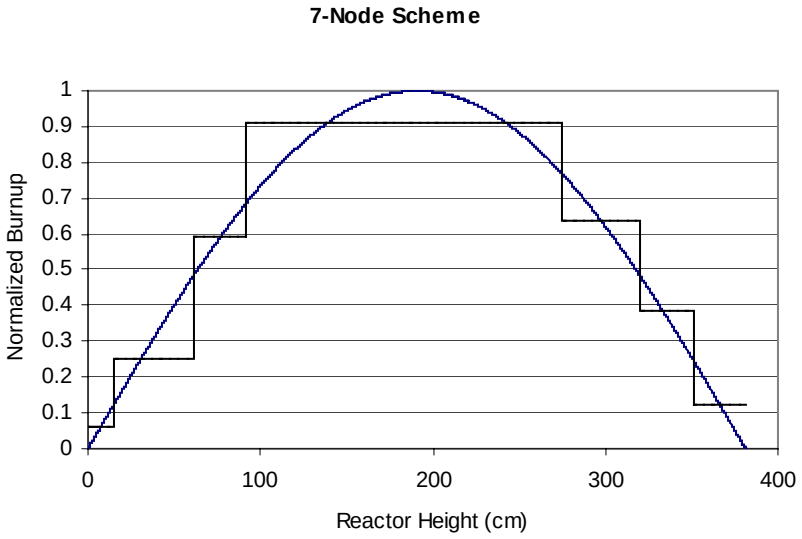

Figure I-5. 7-Node scheme superimposed on an idealized cosine burnup

Source: Spreadsheet Cosine, tab: Graph

A similar problem was encountered when evaluating pressurized water reactor (PWR) fuel. In *Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages.* (DeHart, M.D. 1996, p. 84) after extensive computer simulations the author concluded: "the 7 axial nodes are adequate for the range of enrichments and burnups analyzed both with and without fission products". Table I-1 demonstrates the mathematical errors associated with the compressions. Table I-1 utilizes Equation 2 and calculates the area under the cosine curve corresponding to each step of the three different node schemes and presents the associated errors with each step and the total error. A fuel length of 381.25 cm fuel measured in 15.25 cm increments was selected.

Table I-1. Error Analysis of Node Compressions

Height	Cosine	25-Node		10-Node		7-Node	
(cm)	Area[a]	Area[a]	Error[b]	Area[a]	Error[b]	Area[a]	Error[b]
15.25	0.956926	0.957555	0.066%	0.957555	0.07%	0.957555	0.07%
30.5	2.855685	2.857565	0.066%				
45.75	4.709409	4.712509	0.066%	14.13753	0.59%	14.13753	0.59%
61	6.488863	6.493134	0.066%				
76.25	8.165983	8.171359	0.066%	17.92745	0.26%	17.92745	0.26%
91.5	9.714321	9.720716	0.066%				
106.75	11.10946	11.11677	0.066%				
122	12.32939	12.33751	0.066%	37.01253	0.59%		
137.25	13.35489	13.36368	0.066%				
152.5	14.16976	14.17909	0.066%				
167.75	14.76118	14.77089	0.066%	44.31268	0.59%		
183	15.1198	15.12975	0.066%			166.8172	0.60%
198.25	15.23997	15.25	0.066%				

Table I-1. Error Analysis of Node Compressions

Height	Cosine	25-Node		10-Node		7-Node	
(cm)	Area[a]	Area[a]	Error[b]	Area[a]	Error[b]	Area[a]	Error[b]
213.5	15.1198	15.12975	0.066%	45.38925	0.59%		
228.75	14.76118	14.77089	0.066%				
244	14.16976	14.17909	0.066%				
259.25	13.35489	13.36368	0.066%	40.09103	0.59%		
274.5	12.32939	12.33751	0.066%				
289.75	11.10946	11.11677	0.066%				
305	9.714321	9.720716	0.066%	29.16215	0.59%	29.16215	0.59%
320.25	8.165983	8.171359	0.066%				
335.5	6.488863	6.493134	0.066%	11.2278	0.26%	11.2278	0.26%
350.75	4.709409	4.712509	0.066%				
366	2.855685	2.857565	0.066%	3.822664	0.26%	3.822664	0.26%
381.25	0.9569255	0.957555	0.066%				
Total	242.71129	242.8711	0.066%	244.0406	0.55%	244.0523	0.55%

Source: Spreadsheet Cosine, tab: Graph
Note a: The units on the area under the curve would be (GWd/MTU-cm).
Note b: Error evaluation: (Node Area/ Cosine Area –1).

Notice that there is a small inherent error associated with the presented 25-Node data. A 10-node scheme increases this error; however, the inherent error is less that 1% for all nodes and the total. The simple averaging of the four interior nodes of the 7-node scheme was expected to introduce error. Table I-1 mathematically demonstrates that minimal error (0.01%) is introduced by utilizing the 7-node scheme instead of the 10-node scheme.

Of course, the cosine profile is highly idealized and nuclear reactors have significant shielding to prevent complete leakage of neutrons from the ends of the assembly. Initially reactors were designed to maintain as flat a profile as possible, to gain maximum use of all the available fuel. In this initial design a flat profile, up to the very ends of the assembly was an accepted approximation. However, there were two significant changes to this initial simple design.

First, the initial design assumed the spent nuclear fuel (SNF) was valuable and would be reprocessed. Burn cycles were relatively gentle and fuel was discharged in the range of 20-30 GWd/MTU. The designs also called for moderate enrichment levels in the range of 2-3 wt. % U-235.

In the 70's it became apparent that there wasn't going to be a commercial recycling program and the fuel design switched to a "once-through" burn program, where the fuel design was determined by maximizing the cycle time between refueling. This demand for longer burn times introduced higher enrichments, up to 5 wt. % U-235 and more gadolinium poisons which hold back the U-235 in that fuel section until latter in the burn cycle. Also, fuel pellets became smaller diameter and more rods per assembly were added along with other hardware changes.

Another shift in fuel design came in the 80's when reactor lifetime concerns shifted the fuel design to minimize the irradiation of the reactor vessel. In the early 90's, it was common practice to add six inches of natural uranium pellets to the top and bottom of the assembly. This

is commonly referred to as a “blanket”. Although the term implies a separate fuel zone (as in early test reactors), the natural uranium pellets (enrichment of 0.711 wt.% U-235) are physically in the same rods as the rest of the fuel. Several different enrichment zones are usually present in any given BWR assembly.

All of these changes in nuclear fuel design impact the shape of the burnup profile, in general shifting it from the original flat profile design to a more cosine like profile.

An additional complication in Boiling Water Reactors (BWRs) is the void fraction, or the water density, which also varies with reactor height. Water is a reactor coolant and the moderator slowing the fission neutrons to the thermal range required for fissioning of the U-235 atom. Less moderation at the top of a BWR reactor implies a harder neutron spectrum, which will tend to breed more Pu isotopes from U-238. So not only is the burnup profile varying with the reactor height, but there is also a significant change in the production rates of other fissile isotopes. Modern BWR designs maximize cycle time taking advantage of the Pu production and it is estimated that up to 40% of the power production is from fissioning Pu isotopes.

The impact of the correct profile to the safe handling and storage of spent nuclear fuel is significant. From Figure I-2, the cosine profile would imply the ends of the spent nuclear fuel are essentially fresh nuclear fuels. The addition of light water, in an unsafe geometry, could recreate the conditions in a nuclear reactor and potentially achieve an uncontrolled nuclear reaction, called a ‘Criticality’. Additionally, the top end of a BWR would be a unique concern because it would contain a “mixed oxide”, a combination of U-235 and Pu isotopes.

Since reactor designs do have significant reflection, the ends of the SNF are burned. Therefore, more realistic profiles are required to determine the burnups for use in SNF safety calculations.

While Pressurized Water Reactors (PWRs) can generally be described with a burnup profile, BWRs must be described with moderator density profiles, initial enrichment (and Gd wt. %) profiles and burnup profiles.

This calculation is intended to demonstrate that conservative BWR profiles can be generated from available reactor data. The profiles will allow the user to project individual node burnups and effectively evaluate the reactivity of each node in a BWR assembly based on bounding the available utility data.

ATTACHMENT II

PARAMETRIC STUDY

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ATTACHMENT II PARAMETRIC STUDY

Two parameters affecting SNF have a significant effect on the final k_{eff} of the WP, burnup and BWR moderator/coolant density. To create usable profiles, the effects of the two parameters must be understood. To simplify the problem, each isotopic inventory generated from the SAS2H runs is used for all assemblies in the entire 44-BWR WP for each individual MCNP run. The single material or 'single node' value of k_{eff} demonstrates the effect of varying the individual parameter (burnup or moderator density).

Control blade placement during irradiation in the BWR core also has a significant effect on the final waste package k_{eff}. However, control blade perturbation is outside the scope of this document.

SAS2H cases developed for the LaSalle Unit 1 BWR reactor core were used for this parametric study. The burnups in Table II-1 were used.

Table II-1. Selected Cases and Burnup values

Cycle 1		Cycle 2		Cycle 3		Average
burn (MW)	days	burn (MW)	days	burn (MW)	days	Burnup (GWd/MTU)
41.248	375.32					15.481
75.056	375.32					28.170
35.561	375.32	35.853	467.14			30.095
89.125	375.32					33.450
37.656	375.32	37.965	467.14	38.348	196.09	39.388
46.027	375.32	46.405	467.14	46.873	196.09	48.144

Source: Spreadsheet: md-Test, tab: Time-step

To further simplify the parametric study, all enrichments were adjusted to the highest LaSalle enrichment of 3.65 wt.% U-235 (CRWMS M&O 1999b, Section 4). The gadolinia rods were adjusted to the same 0.73 wt. % natural Gd_2O_3. All burnup cases in Table II-1 were executed at BWR coolant/moderator levels of 0.1, 0.2, 0.3, 0.4, 0.5, 0.6, and 0.7 g/cm^3 H_2O.

The tabulated k_{eff} values are available on the CD under spreadsheet MD-Test, tab keff.

Figure II-1 demonstrates the expected decrease in WP k_{eff} with burnup. Additionally, it clearly shows the effect of varying BWR coolant moderator density.

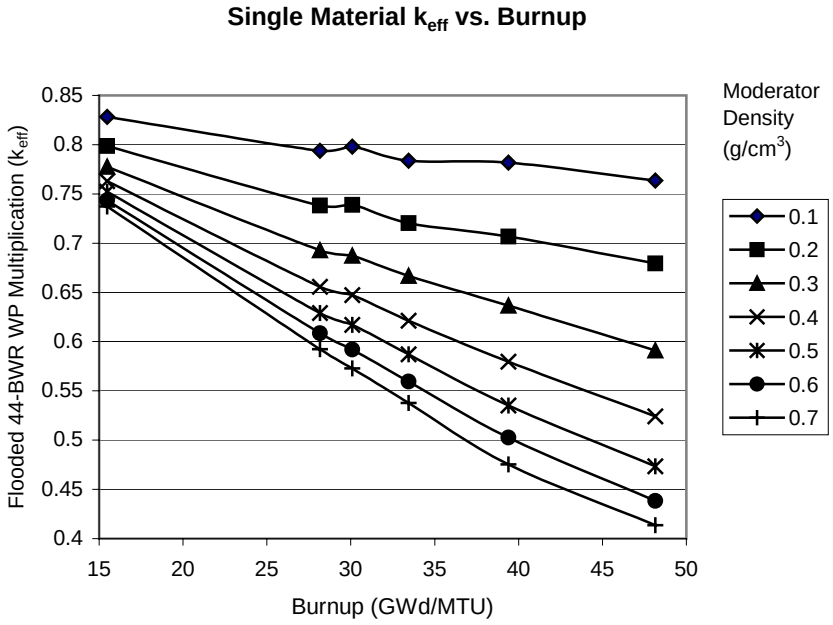

Figure II-1. Single Material k_{eff} with Burnup

Source: Spreadsheet: md-Test, tab: keff

WP k_{eff} is inversely proportional to both burnup and moderator density. The higher the burnup the lower the WP k_{eff}; the higher the moderator density (bottom of the core) the lower the WP k_{eff}. Using the same data points, Figure II-2 shows the relationship between WP k_{eff} and BWR core moderator/coolant density.

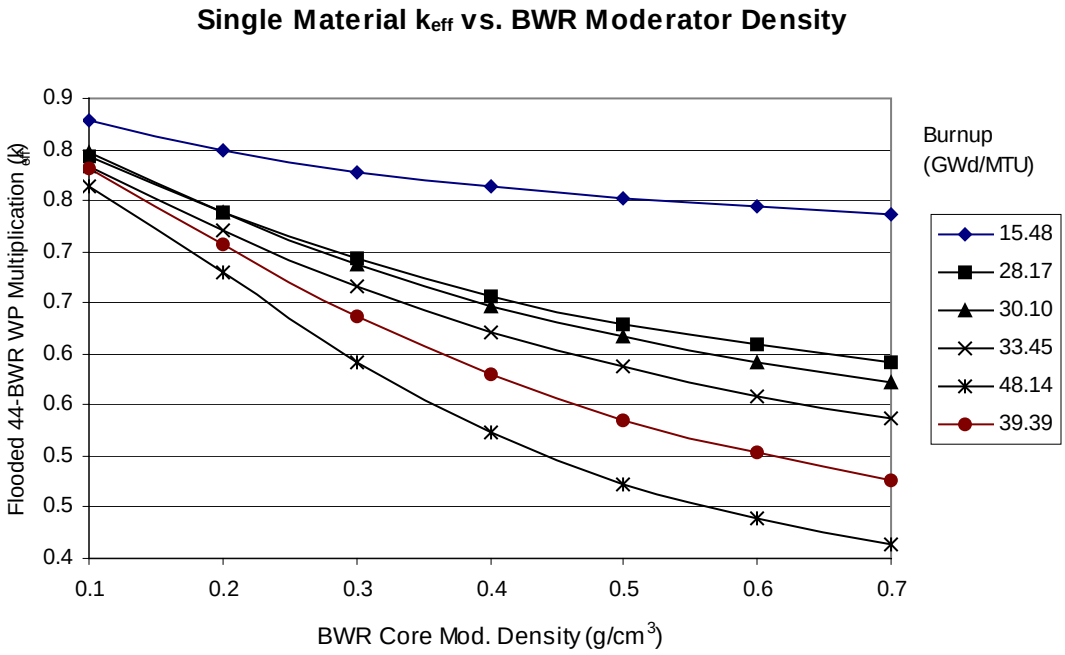

Figure II-2. Single Material k_{eff} with Moderator Density

Source: Spreadsheet: md-Test, tab: keff

From Figure II-2, it is obvious that the lower the moderator density, the higher the WP k_{eff}.

This parametric study demonstrates that the conservative (i.e.: results in the highest WP k_{eff}) burnup is always the minimum value of the burnup. The conservative value of the moderator density is always the minimum value of the moderator density.

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ATTACHMENT III

TESTING CRITERIA

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ATTACHMENT III TESTING CRITERIA

Hypothesis:

The statistically developed burnup and moderator density profiles derived from the available CRC data bases can be used to conservatively predict the effective neutron multiplication factor (k_{eff}) of SNF in flooded 44-BWR waste packages.

Test:

To test the Hypothesis, the following steps are required:

1. A control group of the 25-node CRC data, one (at least) for each of the six burnup profiles.

2. For each of the six assemblies representing a burnup group, each of the 25 individual nodes is processed by SAS2H, to obtain 25 distinct isotopic inventories.

3. A MCNP input deck split into 25 six-inch sections is required, with the 25 isotopic inventories for each of the nodes. Execute a MCNP run for each burnup group with the 25 isotopic inventories.

4. Since the profiles are based on 7-nodes, a 7-node compression of the 25-node CRC data points is required, using the same 7-node layout as developed for the profiles.

5. 7 SAS2H input decks corresponding to the 7-node compression data are required to obtain 7 distinct isotopic inventories.

6. A MCNP input deck split into the 7 axial sections is required. Execute a MCNP run for each burnup group with the 7 isotopic inventories.

7. Develop data representing the same six assemblies from the profiles.

8. 7 SAS2H input decks corresponding to the 7-node profile data are required to obtain 7 distinct isotopic inventories for each assembly.

9. Execute a 7-node profile MCNP deck for each of the six profiles.

10. Repeat the above steps with just the data anticipated to be available with typical BWR assemblies. (i.e.: Using only the highest enrichment with no information on enrichment or gadolinia gradients.)

Note: The test requires a minimum of:
(6 groups * 25 nodes) CRC Detailed + (6*7) CRC Compressed Detailed + (6*7) Profile Detailed = 6*39 = 234 SAS2H runs,

6 CRC Detailed + 6 CRC Compressed Detailed + 6 Profile Detailed = 18 MCNP runs,

(6 groups * 25 nodes) CRC Simplified + (6*7)CRC Compressed Simplified + (6*7) Profile Detailed = 6*39=234 SAS2H, and

6 CRC Simplified + 6 CRC Compressed Simplified + 6 Profile Detailed = 18 MCNP runs,

for a minimum of 468 SAS2H runs feeding 36 MCNP runs.

Passing Criteria:

The following sets of data will be generated:

Table III-1. Detailed Multiplication Table

Burnup Group	25-CRC Detailed	7N-CRC Detailed	Percent Error	7P Detailed	Percent Higher
10-14.9					
15-19.9					
20-24.9					
25-29.9					
30-34.9					
35-40					

Table III-2. Simplified Multiplication Table

Burnup Group	25-CRC Simplified	7N-CRC Simplified	Percent Error	7P Simplified	Percent Higher
10-14.9					
15-19.9					
20-24.9					
25-29.9					
30-34.9					
35-40					

1. The 7-node compressions (detailed and simplified) of the 25-node data should be comparable. Ideally the average error should be less than 1%.

2. The 7-node profile cases **shall** all be above (i.e.: conservative to) the 25-CRC control cases. Additionally, the 7-node profile cases **may** be above the 7-node CRC compressed data.

3. Trends in detailed and simplified cases should be comparable; it is expected that the simplified cases will be higher in multiplication because the uranium enrichment was artificially increased. Trends in burnup (i.e.: smooth lines as seen in the Cooper profile demonstration) should not be expected because each of the burnup cases is a different assembly with unique burn history.

4. The profiles were developed directly from the CRC database using the $T_{95/95}$ minimum values for moderator densities. A parametric study demonstrates a strong increase in k_{eff} with lower moderator density. Additionally, $T_{95/95}$ minimum end burnups were used in the profiles. Therefore, given that ALL of the profile k_{eff}s are above the 25-CRC control cases, the above test cases are considered sufficient to pass the test.

5. In the event that NOT ALL of the 7-node profiles k_{eff}s are conservative to the 25-node CRC, k_{eff}s calculations for more assemblies must be performed. A confidence of 95% would be required for each burnup group profile. (19 conservative assemblies would be required for each non-conservative assembly.) If there is good agreement (less than 1%) between the 25-node CRC and the 7-node CRC compressions, then the 7-node CRC compressions can be used for the additional comparisons.

6. The successful completion of this test demonstrates the viability of the profile approach to bounding SNF. ANS/ANSI 8.1-1998 *"Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors"*, Section 4.2.5 specifically states:

   "In the absence of directly applicable experimental measurements, the limits may be derived from calculations made by a method shown by comparison with experimental data to be valid…"

   Breaking down the Standard's wording for this application: *"direct applicable experimental measurements"* . Direct measurements of the burnup for each section of all BWR SNF assemblies is not available.

   "the limits": For this application, the limits on burnup applied to each fuel section is defined by the burnup profile.

   "derived from calculations": The 7-Node Profile is statistically derived from the large number of assemblies in the CRC database.

   "calculations made by a method shown by comparison with experimental data to be valid…" This test *calculates* the k_{eff} values generated from a flooded BWR WP (accident scenario) utilizing isotopic concentrations from the 7-Node Profile *method* as *compared* to the 25-Node CRC *experimental data* (control cases). The comparison was repeated for six burnup ranges. Successful completion of this test *validates* the *method* of using the 7-Node Profiles.

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ATTACHMENT IV

POISON SELECTION

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ATTACHMENT IV POISON SELECTION

During the development of this calculation, waste package designs utilized a 0.5 cm thick metal plate in the baskets. The metal was specified as Neutronit A978 with a density of 7.76 g/cm^3 as shown in Table 16. The metal contains approximately 0.3 weight percentage boron-10 so it is an effective neutron poison in the thermal ranges of concern. However, due to corrosion concerns, the final design may incorporate a newer metal that utilizes gadolinium.

The new metal is referred to as nickel-gadolinium (Ni-Gd) and specified by ASTM B932-04 (2004). The new alloy is specified to have a density of 8.76 g/cm^3, 11% higher than the density of Neutronit. Table IV-1 was developed from the ASTM standard.

Table IV-1. Alloy B932-04, Ni-Gd

	Composition %		Recommended
Element	**Min (wt.%)**	**Max (wt.%)**	**Av. (wt.%)**
Mo	13.1	16	14.55
Cr	14.5	17.1	15.8
Fe	0	1	0.5
Co	2	2	2
C	0	0.01	0.005
Si	0	0.08	0.04
Mn	0.5	0.5	0.5
P	0.005	0.005	0.005
S	0.005	0.005	0.005
Ni *	67.975	61.185	64.58
O	0.005	0.005	0.005
N	0.01	0.01	0.01
Gd	1.9	2.1	2

Source: ASTM B932-04, 2004, Table 1

Note: * Balance is the remainder of the material.

To further develop Table IV-1 into usable MCNP inputs, natural isotopic percentages were applied to molybdenum, chromium, iron, nickel and gadolinium. The natural isotopic atom percentages were obtained from Parrington et al. (1996). The development tables are available in the spreadsheet: Ni-Gd, tab: Mat.

Additionally, due to guidance from the NRC (NUREG-1567, Section 8.4.1.1) only 75% credit is taken for the gadolinium. Table IV-2 represents MCNP inputs utilizing the chemical composition from Table IV-1 and reducing the metallic gadolinium concentration from 2% to 1.5% per the guidelines.

Table IV-2. MCNP inputs for Alloy B932-04, Ni-Gd

Isotope	MCNP ZAID[a]	Density (g/cm^3)
C-nat	6000.50c	0.0100
Mn-55	25055.50c	0.5000
Si-nat	14000.50c	0.0800
Cr-50	24050.60c	0.6602
Cr-52	24052.60c	13.2247
Cr-53	24053.60c	1.5283
Cr-54	24054.60c	0.3868
Ni-58	28058.60c	43.3679
Ni-60	28060.60c	17.2778
Ni-61	28061.60c	0.7637
Ni-62	28062.60c	2.4717
Ni-64	28064.60c	0.6537
Mo-nat	42000.50c	14.5500
Co-59	27059.50c	2.0000
Gd-152	64152.50c	0.0029
Gd-154	64154.50c	0.0320
Gd-155	64155.50c	0.2187
Gd-156	64156.50c	0.3045
Gd-157	64157.50c	0.2343
Gd-158	64158.50c	0.3742
Gd-160	64160.50c	0.3335
Fe-54	26054.60c	0.0565
Fe-56	26056.60c	0.9190
Fe-57	26057.60c	0.0216
Fe-58	26058.60c	0.0029
S-32	16032.50c	0.0050
P-31	15031.50c	0.0050
O-16	8016.50c	0.0050

Source: Spreadsheet: Ni-Gd, tab: Mat
Note a: ZAID is the MCNP input designation of the isotope

To prove the 7 Node Profiles developed in this calculation are conservative to 25 Node CRC data, Figure 6 was generated. Figure 6 illustrates three cases for all six burnup ranges. The first case is the 25 node control cases, designated as “25-Detailed”. The second set is a straightforward mathematical compression of the 25 node CRC data points to a 7 node layout, designated as “7N-Detailed”. The third set of points represents the profile information, “7P-Detailed” which are more reactive than either the 25 node control cases or its 7 node compression.

For this attachment, the 25 node control cases and the 7 node profiles MCNP cases were rerun. Material number four was replaced with the MCNP inputs for Ni-Gd, Table IV-2. All of the cell cards, which specified material four with a density of 7.76, were changed to 8.76 (g/cm^3).

Similar to Table 40, Table IV-3 demonstrates that the 7 Node Profile data generate conservative k_{eff}s to the 25 Node CRC (Control) assemblies.

Table IV-3. MCNP Results for Ni-Gd Cases

Representing Burnup Group[a]	Actual Burnup Value[a]	CONTROL 25-Detailed	TEST 7P-Detailed	Percent Higher
10-14.9	12.425	0.74692	0.75513	1.10%
15-19.9	17.739	0.71843	0.72536	0.96%
20-24.9	22.938	0.67935	0.72213	6.30%
25-29.9	27.476	0.68282	0.7073	3.59%
30-34.9	32.35	0.65333	0.68023	4.12%
35-40	39.74	0.58613	0.65997	12.60%

Source: Spreadsheet: Ni-Gd, tab 0.15B

Note a: Units of the burnup are in (GWd/MTU).

Similar to Figure 6, Figure IV-1 demonstrates that all of the 7 Node profile WP k_{eff} are conservative to (i.e.: higher) that the 25 Node control cases.

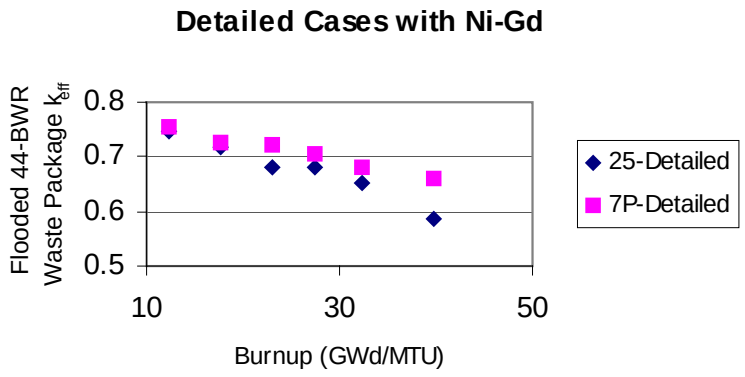

Figure IV-1. Detailed Cases with Ni-Gd

Table IV-3 and Figure IV-1 demonstrate that the 7-Node profiles (as presented in Tables 42-48) can be applied to the Ni-Gd poison plates. Use of the 7-Node Profiles is based on the change in k_{eff} (or delta) between the same WP designs, only changing the SNF isotopic concentrations. Therefore, the methodology for demonstrating 7-Node profiles is insensitive to WP design changes.

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ATTACHMENT V

CHECKING PROCESS

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ATTACHMENT V CHECKING PROCESS

During the checking of this document, two technical issues were brought up. Due to the length of the explanations, they were added in this attachment.

V.1 Significant Digits used in Node Fractions from Table 3.

The node fraction scheme in Table 3 was originally developed utilizing Rev. 00 of *Summary Report of Commercial Reactor Criticality Data for Quad Cities Unit 2*, CRWMS M&O 1998c, Table 4-1. Prior to performing the Cooper Demonstration in Section 5.3 or the extensive LaSalle Test in Section 5.7 it was found that the Quad Cities document had been revised to Rev. 01 as seen in CRWMS M&O 1999a.

The node fractions selected for use in the profiles were taken as the most occurring (mode) lengths from the seven different node fractions sets. The single change from Rev. 00 to Rev. 01 of CRWMS M&O 1999a was to the node lengths presented in Table 4-1. The procedure this calculation used for selecting a common node layout did not change. The most occurring fraction length for each node was updated with the revised Table 4-1 from CRWMS M&O 1999a. The revised node lengths are reported in Table V-1.

Table V-1. Quad Cities Revised Node Lengths

Nodes	A-D	E-M
1(Bottom)	15.24	15.24
2	30.48	30.48
3	30.48	30.48
4	45.72	45.72
5	30.48	30.48
6	45.72	45.72
7	45.72	45.72
8	45.72	45.72
9	60.96	45.72
10 (Top)	15.24	30.48
Total	365.76	365.76

Source: CRWMS M&O 1999a, Table 4-1

When turned into a fractional node scheme, the above data is similar to the other five node schemes, but off by a slight fraction. Table V-2 compares the 4-digit nodes to the ones printed in Table 3 of this calculation.

Table V-2. Accuracy of Grand Gulf Fractional Node Lengths

	4 Digit Accuracy		From Table 3 2-Digit Accuracy	
Nodes	**A-D**	**E-M**	**A-D**	**E-M**
1(Bottom)	0.0417	0.0417	0.04	0.04
2	0.0833	0.0833	0.08	0.08
3	0.0833	0.0833	0.08	0.08
4	0.1250	0.1250	0.13	0.13
5	0.0833	0.0833	0.08	0.08
6	0.1250	0.1250	0.13	0.13
7	0.1250	0.1250	0.13	0.13
8	0.1250	0.1250	0.13	0.13
9	0.1667	0.1250	0.17	0.13
10 (Top)	0.0417	0.0833	0.04	0.08
Total	1.0000	1.0000	1.00	1.00

Source: Spreadsheet abc, tab: time

The most occurring fractional length for Node 3 from all seven layouts is 0.0833, while rounded to 0.08 in Table 3.

During checks of rod powers, it was noticed that the average rod power was not balancing. The problem was traced to the node values, which seemed to sum to 1.00 but were actually summing to 1.0033 due to the odd fraction for Node 3 from the Grand Gulf revision.

Simply normalizing the node values to 1.0000 corrected the power levels, as shown in Table V-3.

Table V-3. Most Occurring (Mode) Values Selected for use with the Profile

Nodes	Mode	Normalized Mode 4-Digit Accuracy	2-Digit Accuracy used in Tables 3, 12, 23-24 & 26-31
1(Bottom)	0.0400	0.0399	0.04
2	0.1200	0.1196	0.12
3	0.0833	0.0830	0.08
4	0.1200	0.1196	0.12
5	0.1200	0.1196	0.12
6	0.1200	0.1196	0.12
7	0.1200	0.1196	0.12
8	0.1200	0.1196	0.12
9	0.0800	0.0797	0.08
10 (Top)	0.0800	0.0797	0.08
Total	1.0033	1.0000	1.00

Source: Spreadsheet abc, tab: time

Notice that the worksheets, tables 42-47, instruct the analyst to double check this by multiplying the % length by the calculated power per node, under the header Check X. The analyst is instructed to sum the check column. This procedure verifies that the original average rod burnup is returned.

In Table 12, it is reported that Nodes 4-7 have a fraction 0.48 times a power level of 22.2858 (GW) resulting in a burn of 10.662 (GWd/MTU). However, note that Nodes 4-7 would more accurately be the product of 0.1196 times four nodes which yields 0.47842. 0.47842 times a power level of 22.2858 does correctly yield 10.662 (GWd/MTU).

V.2 MCNP Modeling of Cooper 7x7 Fuel Channels

During the extensive checking of this document, a slight geometry overlap was indicated in the MCNP representations of the channel corners used in the 7x7 Cooper Assemblies. This overlap is displayed graphically as a red dotted line in the cusp of these corners. This coding was used in Section 5.3 "Profile Demonstration". An indicated geometry overlap raises the potential for some material being inadvertently clipped out of the model. The adjustments in Table V-4 were made to assure that material was not clipped out of the 7x7 channel.

Table V-4. Channel Corner

Surfaces	Geometry	Radius	Comment
180-183	C/Z	1.04	$ Channel Outer Corner Radius *Radius Mod.*
184-187	C/Z	0.723	$ Channel Inner Corner Radius " "
Radii were adjusted to:			
180-183	C/Z	1.01	$ Channel Outer Corner Radius *Radius Mod.*
184-187	C/Z	0.72	$ Channel Inner Corner Radius " "

Source: Spreadsheet: Check, tab: Check

The adjustment in Table V-4 stopped the geometry processor from reporting an interference. However, it was judged slight enough not to warrant rerunning all of the codes in the section. To verify this, two copies of file P-32-4 were made. Both copies had the material processor callout "Print –128" to obtain more detailed material information. File "test" had radius adjustments from Table V-4; file "testb" kept the same radii as the original file P-32-4.

There was a slight difference noted in the resulting k_{eff} data, as show in Table V-5.

Table V-5. k_{eff} Comparison

Note	file	k_{eff}	Sigma
Original P-32-4	testb	0.69565	0.00064
Adjusted corner	test	0.69472	0.00065
Percent difference:		0.13%	

Source: Spreadsheet: Check, tab: Check

Table V-5 demonstrates that the change is within two values of sigma, the inherent error associated with Monte Carlo numerical calculations.

Furthermore, the mass determination performed by the code for the channel material was unchanged as demonstrated in Table V-6.

Table V-6. Mass Comparison

File/Mat. #	atom density	gram density	input volume	mass	pieces
test					
50	290	4.34E-02	6.55E+00	1.57E+02	1.03E+03
testb original					
50	290	4.34E-02	6.55E+00	1.57E+02	1.03E+03

Source: Spreadsheet: Check, tab: Check

Therefore, there is no significant impact from the potential geometrical interference noted in the demonstration runs in Section 5.3.

ATTACHMENT VI

CD FILE DESCRIPTION

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ATTACHMENT VI CD FILE DESCRIPTION

This attachment contains a listing and description of the zip file contained on the attached CD of this calculation (Attachment VII). The CD was written using the Roxio Direct CD Creator Version 5.2.0.56 with model AL-DT-ST RW/ DVD GCC-424ON CD-rewritable drive for personal computers, and the zip archive was created using WINZIP 8.1.

The following is a description of the archive file. Table VI-1 provides the directory structure and number of files in the archive. The SAS2H Cases section provides a description of the SAS2H files that were retained. The input conversions performed with spreadsheets are described in Section VI.3. The MCNP cases are explained in Section VI.4.

Filename	# of zipped files	File Size (bytes)	File Date	File Time	Description
CAL-DSU-NU-000005.zip	17,547	150,921,388	9/23/2004	2:12 PM	Zipped Archive containing Microsoft Excel spreadsheets, SCALE and MCNP inputs.

Table VI-1. CD Directory Structure

Directory	Files
Main	5
P-12-3	49
P-12-4	49
P-12-5	49
P-17-3	49
P-17-4	49
P-17-5	49
P-22-3	49
P-22-4	49
P-22-5	49
P-27-3	49
P-27-4	49
P-27-5	49
P-32-3	49
P-32-4	49
P-32-5	49
P-37-3	49
P-37-4	49
P-37-5	49
P-mcnp	36
Profile-Isos	20
LaSalle-TEST	see Table VI-3
Attachments	Four Sub-Directories See Section VI..6

VI.1 PROFILE DEVELOPMENT: MICROSOFT EXCEL SPREADSHEETS

A spreadsheet was developed for each CRC data set available: Grand Gulf Unit 1 (Punatar 2001), Quad Cities Unit 2 (CRWMS M&O 1999a) and LaSalle Unit 1 (CRWMS M&O 1999b). The compilations of the three CRC profiles are done in GGTH-Node. There are extensive links between the files, which require the same tree directory and file names to remain functioning. Each spreadsheet contains multiple work sheets broken down as:

Each of the following tabs is located in all three main CRC Spreadsheets. The main CRC spreadsheets are in the Main Directory.

Data: This is the data from the source documents.

Profile-All: This sheet derives the burnup profiles for each statepoint available in the Data worksheet.

Sort*: The burnup profiles from "Profile-All" were copied and pasted here as text. The profiles are then sorted by burnup and broken into the six burnup value ranges. The spreadsheet finds the average and minimum burnup values for each node utilizing built-in functions. Traceability of each point to the original data point is preserved. An individual sort sheet is done for each rod layout; a listing of all rod layouts is presented in Table 3.

Temp: A take off of the available temperatures from the data sheet.

Temp-Sort: Sorting the temperatures in the same groupings used in the profiles.

Mod or **ModDen:** A take off of the available moderator densities from the data sheet.

Mod-Sort or **ModDen-SortB:** Sorting the moderator densities in the same groupings used in the profiles.

Nodes or **Node Wts**.: Breaks down the Rod layout scheme for the different fuel rods.

In spreadsheet QCTH-2, a worksheet is added to assist the reader:

Demo: Demonstrates the algebra used to generate the average burnup and the burnup profile.

In addition, spreadsheet GGTH-node contains these worksheets linking all three of the spreadsheets:

Nodes: Provides the Node lengths of all the available data, finds percentages and the mode percentages that are used as representative.

Profile-Comp: Compresses all of the available burnup data into one set of burnup profiles.

Temp-Comp: Compresses all of the available fuel temperature data into one set of fuel temperature profiles.

Mod-Comp: Compresses all of the available moderator density data into one set of moderator density profiles.

Odds: Calculates the overall odds presented in Tables 49 & 50.

Other Files:

readme.doc: A Microsoft Word copy of this Attachment and further user instructions.

temp.xls: A spreadsheet that generates the temperature profiles.

VI.2 SAS2H CASES

In the individual directories, labeled P-*-3, 4 and 5 the "P" designates profile, the "*" is the whole number part of the burnup value (12.5, 17.5…37.5 GWd/MTU). The 3, 4 and 5 represent the initial fuel assay in wt % U-235.

N.inp* files are the SAS2H input files, where N* represents the corresponding fuel node.

N.log* files contains an echo of the input and pertinent information extracted from the SAS2H output file prior to discarding, to indicate that the case ran successfully. The N* represents the corresponding fuel node. (All safety related calculations retained log files, however, the parametric study did not retain the log files)

N.msgs* files are generated by SAS2H and contain the standard run-time messages associated with the SAS2H calculations, where the N* represents the corresponding fuel node.

*ft72f001.N** files are binary files generated by ORIGEN-S for each time step, which were retained, that contain the isotopic concentrations as a function of time. The N* represents the corresponding fuel node.

act_N.mass* files contain the extracted actinide isotopes from the *ft72f001.N** files and provides them in units of grams.

fp_N.mass files* contain the extracted fission product isotopes from the *ft72f001.N** files and provides them in units of grams.

le_N.mass files* contain the extracted light element isotopes from the *ft72f001.N** files and provides them in units of grams, (light elements, including residual Gd-155, B-10 that were not consumed by the irradiation process).

One example output file is included, *N04.output* in file P-37-5. The maximum burnup and flux is reported in Table 11.

VI.3 DEMONSTRATION INPUT CONVERSION: MICROSOFT EXCEL SPREADSHEETS

Directory: Profile-Isos

P-*-3, 4, &5.xls: Each individual MCNP run has a corresponding spreadsheet that complies the seven SAS2H runs into a material input deck. Each individual spreadsheet has seven worksheets N01-N10, which have copied into them the SAS2H outputs from fp_N*, act.N*, and le_N*.mass. The data is then pulled into an algorithm that holds the oxygen

content fixed to the initial UO_2 fuel and scales the isotopic concentrations for use with MCNP. The worksheet, "comp" does a quick comparison of the isotopic concentrations as a double check. The Worksheet "MCNP-Run" is a cut and pastable version that was used to input the data to MCNP decks.

keff.xls: Compiles the resulting k_{eff} information from the mcnp codes and produces the graphics.

abc.xls: abc is a set of additional worksheets that execute common tasks:

- Mass - calculate the length of rods required for 1 MTU,
- Time - sets up the artificial time steps for the SAS2H codes
- Assay - Given the U-235 assay of fresh fuel, Assay provides estimates of U-234, U-236, and U-238.
- MCNP - Works out the MCNP input deck for splitting the fuel into 7 sections

VI.4 DEMONSTRATION MCNP CASES

The single directory P-mcnp contains the MCNP input files (P-*-3, 4, or 5 – see below), and the MCNP output files with the same name followed by an "o". Six burnup levels were executed at three initial enrichments resulting in 18 MCNP runs.

(P--3, 4, or 5) "P" stands for profile case, the asterisk (*) represents the whole number of the burnup case 12 for 12.5, 17 for 17.5 up to 37 for 37.5 GWd/MTU. The 3, 4 or 5 represents the weight percent U-235, or enrichment.

VI.5 LASALLE TEST CASES

Under the directory LaSalle-Test, there are three sub-directories: SAS2H-25N, SAS2H-7N, and SAS2H-7P. The SAS2H-25N directory contains all of the files associated with the 25 Node control CRC cases. The SAS2H-7N directory contains all of the files associated with the 7-node CRC compressed cases. The SAS2H-7P directory contains all of the files associated with the 7-node profile or test cases.

Each of the three sub-directories contains six sub-directories: LS-12 through LS-39, a subdirectory for each burnup group. Each also contains a MCNP directory, where all of the MCNP cases for those burnup groups are contained. SAS2H-7N subdirectory also contains an additional MCNP subdirectory labeled: Mod-Den-MCNP, this sub-directory holds all of the MCNP input and output files utilized for the parametric study, Attachment II.

Each of the burnup group sub-directories LS-12 through LS-39 contains two sub-directories: Detailed and No-Gd. SAS2H-7N also contains a Mod-Den directory for the parametric study (see Table VI-3). Each of these sub-directories contains all of the SAS2H files with the naming convention described in Section VI.2. Additionally, they contain multiple ORGN-case*.MCNP files, which are isotopic databases each cut/paste MCNP ready. The number represented by the asterisk distinguishes the age of the isotopic inventory starting from the end of the last reactor cycle. (Usually, the SAS2H specifies a 5-year waiting period; however, the last cooling period is not used) The Table VI-2 matches the years of cooling to the code number.

Table VI-2 Available Years Aged from SAS2H Outputs

Case #	years aged	Case #	years aged	Case #	years aged
1	0yr	10	10K	19	22.5K
2	5yr	11	11K	20	25K
3	10yr	12	12K	21	27.5K
4	33yr	13	13K	22	30K
5	100yr	14	14K	23	35K
6	333yr	15	15K	24	50K
7	1K	16	16K	25	75K
8	3.333K	17	18K	26	100K
9	5K	18	20K		

Source: Spreadsheet LaSalle-Runs, tab: Years

Additionally, the burnup groups in the SAS2H-7N directory contain Mod-Den directory, which contain the SAS2H input files and associated run files. The log files for these were not included.

Table VI-3 LaSalle Test File Structure

Directory	Sub1	Sub2	Sub3
LaSalle-Test	SAS2H-25N SAS2H-7P	LS-12	Detailed No-Gd
		LS-18	
		LS-23	
		LS-27	
		LS-32	
		LS-39	
		MCNP	
	SAS2H-7N	LS-12	Detailed Mod-Den No-Gd
		LS-18	
		LS-23	
		LS-27	
		LS-32	
		LS-39	
		MCNP	
		Mod-Den-MCNP	
Total:	3	22	45

These Microsoft Excel spreadsheets are in the LaSalle-TEST directory.

gad-calc.xls: Calculates the weight percent gadolinia in a fuel assembly. Section 5.6, Table 34.

LaSalle-Runs.xls Development and results of the LaSalle test, Section 5.6.

VI.6 ATTACHMENTS

An individual directory was added for Attachments with the following sub-directories:

Sub-Directory: Att.I

Cosine.xls: Graphs an example cosine profile, graphically overlays the 25, 10 and 7-Node schemes. Calculates the error introduced from the approximations, Attachment I.

Sub-Directory: Att. II
md-test.xls: Provides the results and graphs of the moderator test, Attachment II.

(Note Attachment III did not require a spreadsheet or calculations)

Sub-Directory: Att. IV

Ni-Gd.xls: Provides the development of the Nickel Gadolinium plate material and develops graphs with the k_{eff} results.

MCNP Cases:

25*-NG , where , the asterisk (*) represents the whole number of the burnup case 12 for 12.5, 17 for 17.5 up to 37 for 37.5 GWd/MTU. the “25” represents the 25 Node or Control case. Files ending with “NGo” are the output files. File 25-res is a listing of the resulting k_{eff} data.

7P*-NG, similar to the above cases, with the “7P” representing the 7 Node Profile case to be demonstrated conservative to the above control cases.

Sub-Directory: Att. V

Check.xls: Provides detailed geometry data on the channel corners.

MCNP runs: “testb” re-executes MCNP file P-32-4, with an additional “PRINT-128” command for more detailed geometry output.

File “test” only makes minor changes to channel corners as indicated in Table V-4.

Files “testo” and “testbo” are the output files. File “test-res” is a listing of the k_{eff} results.