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MAY 12 1994

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United States Nuclear Regulatory Commission
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Washington, DC 20555

Gentlemen:

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REACTOR PHYSICS METHOD TOPICAL REPORT
SALEM GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

Attached is PSE&G's response to NRC's request for additional information on Salem's Reactor Physics Methods report, NFU-0039, Revision 2. The questions were transmitted via NRC letter dated February 24, 1994.

As a result of our ongoing model validation process, additional benchmarking of the extended burnup model has been completed. Since the submittal of NFU-0039, Revision 2, three cycles of operation have been completed, each of which utilized multiple burnable poison loadings. We have included the results of these three recent cycles together with the five cycles presented in NFU-0039, Rev. 2, to evaluate the biases and reliability factors previously reported.

The results of this current benchmarking show consistent model performance with increased burnup, thus indicating that the NFU-0039, Revision 2 reliability factors remain applicable for the multiple burnable absorber loading and extended burnup designs, which are part of the present Salem fuel management strategy. However, based on the recent benchmarking results and recent EPRI information, biases for Doppler defect and temperature coefficient have been revised, as discussed in the attachment.

200060

A001

Should you have any questions regarding this information, please contact us.

Sincerely,

Attachment
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Responses to the NRC Request for Additional Information
PSE&G Salem Reactor Physics Methods Topical Report,
NFU-0039, Revision 2

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BACKGROUND

A brief background discussion of the PSE&G Salem core physics design and analysis process is provided, prior to responding to the specific requests, to support the explanation of how changes in fuel designs or changes in computer codes are addressed at PSE&G. The paragraphs that follow describe the fuel designs in use at Salem, summarize the methodology change that occurred to support the new designs, describe the PSE&G plans for future Salem cores, and summarize the overall process employed at PSE&G to deal with design and methodology changes. This discussion provides background information relevant to PSE&G's specific responses that follow.

The PSE&G design and analysis process deals with the reality that fuel designs change over time and the computer codes or methodology also change to support the new designs. Model benchmarking or validation is an on-going process. Each cycle, reactivity and power distribution comparisons are made to verify that the model accuracy is within expectations and to determine if any trends in accuracy are developing which may impact future predictions. PSE&G is cognizant of and participates in industry methodology development programs. Changes in fuel design for Salem are gradual. The table below shows the Salem cycles and the trends in burnable poison usage and the discharge burnup.

Unit/ Cycle	Begin Date	Feed U_{235} w%	BOC Burnup (MWD/MTU)	Cycle Burnup (MWD/MTU)	Lead Assembly Discharge Burnup	#BPRs Loaded	#IFBA Pins Loaded
1/7	10/87	3.8	10,620	16,805	40,000	1576	0
1/8	2/88	3.8	11,210	13,766	40,000	1664	0
2/5	11/88	3.6/4.0	11,700	14,614	42,000	1728	0
1/9	6/89	3.6/4.0	11,420	16,521	45,000	1344	0
2/6	6/90	3.8/4.0	12,040	15,994	47,000	1632	0
1/10	4/91	4.0	16,750	11,818	44,000	512	3072
2/7	4/92	3.8/4.0	16,580	9,540	46,000	544	2560
1/11	8/92	4.0/4.4	14,670	13,035	46,000	368	6400
2/8	6/93	4.0/4.4	14,610	16,900*	48,000*	418	5088
1/12	1/94	4.0/4.4	16,055	16,000*	46,000*	512	3072 (1.5X)

* denotes projected values for current cycles.

The change from the EPRI CELL based model to the PSCPM based model was necessitated by the need to utilize multiple burnable poisons for the extended burnup cores. The new fuel design and modeling technique were evaluated by performing sensitivity studies. This established the impact of design decisions or modeling assumptions. Next, the methodology to be used was validated by performing comparative calculations to other codes. This establishes the confidence in the ability to predict the behavior of new designs before they are loaded into the reactor for the first time. Next, benchmarking was performed to prior operating data to compare the accuracy of the old model (CELL based) vs the new (PSCPM based). As changes to the fuel designs were utilized in Salem core reloads, on-going benchmarking was performed.

Salem 2 Cycle 9 will begin operation in the fall of 1994 and contingent upon approval of the licensing amendment request, the fuel will utilize several high burnup design features. Included among these features are the use of Zirlo cladding, an increase in the plenum volume for fission gases, intermediate flow mixers, and a 50% higher ZrB_2 linear loading (1.5X design). The 1.5X IFBA design was loaded for the first time in Salem 1 Cycle 12 at the beginning of 1994. The 1.0X IFBA design was loaded starting with Salem 1 Cycle 10 in 1991.

In summary, the change to new designs is a methodical decision, and the changes in computer codes and the performance of all engineering analysis is controlled and documented per the PSE&G QA procedures. As illustrated in the background discussion above, the PSE&G process is on-going and consists of; 1) sensitivity studies to evaluate the design or methodology change, 2) calculations to validate the performance of the analysis system, and 3) benchmarking to operating data. In step one of the process, sensitivity calculations are performed to determine the impact of either design variables (i.e.; # of IFBA/assembly) or modeling assumptions (i.e.; IFBA coating smeared in clad region or treated explicitly). In step two, predictive calculations are performed and compared to other independent methods (i.e., another code or the fuel vendor results). In step three, measured versus predicted comparisons are made and the model biases and reliability factors are revised as appropriate.

As additional improvements are made to the EPRI-CPM code (current EPRI CPM-3 project), these improvements will be evaluated for incorporation into PSCPM using the above process. Any future methodology changes will be reviewed to determine if the Salem Reactor Physics Methods topical report needs to be revised.

QUESTIONS and RESPONSES

Note that in the questions and responses that follow, all reference numbers refer to the Salem topical report page 5-1 and 5-2 except for questions 1, 8 and 12. For questions 1 and 12, the list of references are the Doppler references found in the response to question 12. The response to Question 8 contains additional new references which are listed at the end of that response.

1 *Provide a detailed discussion of the basis for the PSCPM bias in the Doppler coefficient. Have the Reference 12 and 13 benchmarking comparisons been made with the latest version of PSCPM? Discuss the applicability of the PSCPM Doppler bias and reliability factors and the calculations of References 12 and 13 to the intended Salem fuel designs.*

The basis for the Doppler bias and uncertainty is the EPRI research documented in References 1 and 2 (see Question 12 response for list of Doppler references). The Reference 12 and 13 (of Salem topical report) benchmarking comparisons were made with CPM2 which calculates the exact same Doppler as PSCPM. The impact on the Doppler uncertainty due to different H_2O/UO_2 ratios is shown in Reference 2 and based on this information it is concluded that the bias and uncertainty derived in Reference 2 are applicable to the current and the planned Salem fuel designs. A detailed explanation of these responses is provided in the text that follows.

The original basis for the bias and uncertainty in the PSCPM Doppler coefficient and defect was taken from the investigation of a EPRI project as reported in Reference 1 (see Question 12 response for list of Doppler references). Reference 1 shows that CPM2 consistently gives a higher value for Doppler worth than EPRI-CELL2/Standard Library (CELL2/STD). Pertinent data in Reference 1 shows :

1. The difference between CPM2 and CELL2/STD Doppler worth values for the 3.9% enrichment is approximately 30% at E=O, approximately 20% at E=14 GWD/MT and approximately 18% at E=32 GWD/MT.

2. There is some dependence on enrichment; as the enrichment increases the difference between CPM2 and CELL2/STD decreases.

At the time Revision 2 of the Salem Reactor Physics Methods was completed, the available information incorporated in Reference 1 was the Hellstrand experiments (Reference 4), and some in reactor measurements (References 3, 5, 6). Based on this, it was concluded that CELL2/STD results were the most accurate, and CELL2/STD was taken as the standard. As explained on page 4-10 of the Salem topical, the CPM2 Doppler worths are bounding on the high side as calculated by the code (CPM2 over

predicts Doppler). It was postulated that when the CPM2 values are multiplied by .80 they are bounding on the low side. For application to Salem, the typical average core exposure at BOC was taken to be approximately 14 GWD/MT and at EOC, approximately 32 GWD/MT (Reference 1 shows the difference between CPM2 and CELL2/STD is nearly constant at higher burnup). The data in the table on page 4-10 of the PSE&G Salem topical is from Reference 1, and the factor of .8 makes CPM2 values smaller than CELL2/STD for the exposures of interest. Since the Doppler results from CPM2 and PSCPM are exactly the same, the conclusions drawn for CPM2 are also true for PSCPM. Thus the derivation of the 10% bias and 10% uncertainty on page 4-10.

In July 1993, Reference 2 was published and presented new information. Reference 2 contains detailed comparisons of the various cross section generators with Hellstrand's measurements, Monte Carlo analyses and reactor trends. Reference 2 showed that agreeing with Hellstrand results is not, by itself, a sufficient determination of the accuracy of the Doppler calculation in a reactor lattice. It also showed that References 7 and 8 herein, and Reference 13 in the Salem Reactor Physics Methods report have data that are erroneous. These cases were recalculated and reported in Reference 2. Reference 2 concludes that all the results show CELL2/STD to be giving the best estimate of the Doppler worth. Quantitatively, CELL2/STD was within 2 to 3% of the Hellstrand nominal values, and within 3% of the Monte Carlo nominal values for the enrichments of interest. This supports the earlier Reference 1 approach of taking CELL2/STD as the standard. The later results in Reference 2 showed CPM2 to be 10% higher than the Hellstrand nominal value, and approximately 14% higher than the Monte Carlo nominal value for the mixed oxide case (i.e., the case emulating extended exposure). Reference 2 concludes that for CPM2 a bias of -25% with an uncertainty of $\pm$ 10% is appropriate for fresh fuel, and a bias of -15% with an uncertainty of $\pm$ 10% is appropriate for fuel at 32 GWD/MTU.

Based on the more recent results, the Doppler bias and reliability factor shown in the Salem topical report Table 4.0.1 will be replaced with the following:

bias:
$CAE \leq 32$ bias = -(25% - .3125 * CAE)
$CAE > 32$ bias = -15%
where CAE = Core Average Exposure in GWD/MTU
reliability factor (RFDC) = 10%
reliability factor (RFDD) = 10%

For example; if PSCPM/TRINODE calculated a Doppler coefficient of -2 pcm/F at a BOC core average exposure of 14 GWD/MTU, then the least negative Doppler coefficient would be :

= nominal value * (1 + bias/100) * (1 - RFDC/100)

= -2.0 * [1 - (25 - 0.3125 * 14)/100] * (1 - 10/100)

= -2.0 * .794 * .9 = -1.43 pcm/F

If PSCPM/TRINODE calculated a Doppler coefficient of -4 pcm/F at a EOC core average exposure of 32 GWD/MTU, then the most negative Doppler coefficient would be:

= nominal value * (1 + bias/100) * (1 + RFDC/100)

= -4.0 * [1 - (25 - 0.3125 * 32)/100] * (1 + 10/100)

= -4.0 * 0.85 * 1.1 = -3.74 pcm/F

In the evaluation of Doppler accuracy, the most important physical parameters are the moderator/fuel ratio and the pellet size. The biases and uncertainties determined based on the Reference 2 material are applicable to the current and planned designs as illustrated by the following table:

	Typical Salem Values	Monte Carlo Comparisons (Ref 2)	Pin Cell Comparisons (Ref 1)
Pellet O.R.(cm)	.4186	.39306	.41169
Clad O.R. (cm)	.4759	.45802	.47587
Pitch	1.2649	1.26209	1.25984
$H_20/U0_2$	1.614	1.924	1.645

Table 4-12 in Reference 2 showed that a change in the H_2O/UO_2 ratio from 1.65 to 1.92 changes the difference between CPM2 and CELL2/STD U-238 absorption fraction and hence Doppler results by approximately 1%. The difference in the fuel pin size and pitch for Salem compared to the Reference 1 and 2 data will have little affect on the results and hence the results of Reference 1 and Reference 2 are applicable to Salem fuel designs.

2 *Discuss in detail the changes introduced into the more recent EPRI-CPM2 and PSCPM versions of CPM. What changes have been introduced in PSCPM to calculate the integral fuel burnable absorber fuel designs and mixed core loadings? Are these changes included in the extended burnup model benchmark comparisons and, if not, how will these changes be validated?*

Revision 1 of the Salem topical utilized a CELL based model in which the original EPRI-CPM code was part of the EPRI ARMP methodology (Reference 1). For Revision 2 of the topical, the starting point for the extended burnup model was the EPRI-CPM2 (Reference 11) code and library. Since the EPRI release of CPM2, no modifications to the PSE&G copy of the EPRI-CPM2 code or library were made to incorporate EPRI sponsored modifications. (Note: Changes have been made to CPM2 by the EPRI maintenance group mainly for BWR applications, but these have not been incorporated into the PSE&G version of CPM2).

The CPM2 code as released by EPRI was modified by PSE&G to write punch files for the downstream linking codes and for the explicit treatment of IFBAs. The modified code was named PSCPM. The linking code modifications did not change any of the CPM2 computations; the original Punch option was extended to include data needed in the PDQ and TRINODE input. The PLINK code was written to process the CPM2 calculated items such as; reaction rates, number densities, macro cross sections, fluxes, and geometry data to create the PDQ input files. The BLINK code was written to process the CPM2 calculated items such as; K^{∞}, M^2, κ, Σ_f, $\nu\Sigma_f$, and σ_{xe} to create the TRINODE input files. Explicit treatment of the IFBA design was achieved by modifying CPM2 to permit an additional annulus in the fuel pin. The annulus added is outside of the pellet and is the ZrB_2 coating. As outlined in the background discussion, the new methodology was validated by performing sensitivity studies to evaluate modeling assumptions and by performing comparisons to other computer codes. For the IFBA design, the assumption of an explicit ZrB_2 region vs smearing the ZrB_2 in the clad region was evaluated and pincell results were compared to EPRI-CPM (which permits four annuli in the fuel pin) and EPRI-CELL (both codes described in Reference 1). The new IFBA capability combined with the existing CPM2 capability for burnable absorber rods provides capability for modeling mixed assembly designs containing both IFBAs and burnable absorber rod inserts. The new code, PSCPM, was then used to generate the PDQ and TRINODE inputs which were then used to perform the extended burnup comparisons presented in Revision 2 of the Salem topical report.

For future applications, PSCPM is being converted to execute on RISC based processors and will be designated CPM-2/WSE. This version is functionally equivalent to PSCPM for Salem reactor physics analysis and produces identical results.

3 *Describe in detail the changes made in TRINODE including the treatment of burnable absorber reactivity input.*

A comprehensive list of the changes made to TRINODE since Revision 1 of the Salem topical is contained in Table 1. A summary of the key changes for the extended burnup model is presented below.

The TRINODE code was modified to accept table input instead of curve fits and to explicitly account for the multiple burnable poison designs. Prior to modifying TRINODE, PSCPM calculations were performed for various multiple burnable poison designs to evaluate each of the reactivity effects modeled in TRINODE. Based on this examination the traditional "B-constants" of EPRI-NODE-P (Reference 1) were replaced with explicit tables and the functional dependence of the independent variables was extended to multiple dependent parameters. For example, $\Delta\rho_{exposure}$ vs burnup was replaced by tables of K^{∞} as a function of burnup, instantaneous moderator temperature, and moderator temperature history. The tables are input to TRINODE by fuel type. A fuel type in TRINODE is a unique plane of an assembly based on its fuel dimensions, U_{235} enrichment, number of BPR rodlets, and number of IFBA pins. Thus the reactivity of the newer fuel designs is explicitly modeled as a unique fuel type.

4 *Describe the fuel designs (e.g., multiple burnable absorber and integral fuel burnable absorber) to which the PSE&G methodology will be applied. Discuss the adequacy of the Chapter 4 benchmarking for qualifying the methodology for these applications.*

Salem fuel is a 17x17 lattice which may contain 0 to 24 Pyrex Burnable Poison rodlets inserted into the guide tube locations. The number of fuel pins which may contain the ZrB_2 coating ranges from 0 to 164. The typical designs currently utilized are 64 or 104 IFBA with 0 to 12 BPRs. Occasionally some cycles may contain assemblies with inert rods, steel or zirc, which are used as filler rods to replace damaged fuel pins following fuel reconstitution. Salem 1 cycle 11 and Salem 2 cycle 8 are recent examples of this.

As explained in the introductory background text, the latest design utilizes a 1.5X boron loading in the IFBA pins. Benchmarking is an on-going process at PSE&G. When Revision 2 of the Salem topical was submitted, the report contained the most recently

completed cycles for each reactor unit. Since that time, Salem 1 cycles 10 and 11 and Salem 2 cycle 7 have completed operations. All three of these cycles contain multiple burnable poisons and have been benchmarked as part of the on-going model validation. Attachment 1 to this response presents the latest results.

PSE&G has no current plans to utilize fuel designs other than those that are uranium fuel with boron as the burnable poison. Any future design changes would be evaluated and the methodology revised if warranted following the process outlined in the background discussion.

5 *How do the Revision 2 rod worth standard deviation and reliability factor based on calculation to measurement comparisons compare with the Revision 1 values?*

Revision 2 values compared to the Revision 1 values (reported on page 3 - 4 of Physics Methods Report) are 60 pcm vs. 91 pcm reliability factor for rods worth less than 600 pcm, a 9% vs. a 12% reliability factor for rods worth more than 600 pcm, and a 8% vs. a 16% reliability factor for total rod worth. In addition, each of the Revision 2 factors is less than the the extended burnup model reliability factors tabulated in Table 4.0.1, page 4 -2.

6 *In Table 4.2.1, what is causing the large difference between the Revision 1 and Revision 2 predictions of the Cycle 8 Salem 1 isothermal temperature coefficient?*

The extended burnup model contains explicit tables of K^{∞} as a function of burnup, instantaneous moderator temperature, and moderator temperature history. This treatment of moderator temperature and history is a significant change from the original CELL based model. The original CELL based model used a BOL K^{∞} vs moderator temperature and a $\Delta\rho_{exposure}$ vs burnup at the average moderator temperature. Also, the Doppler component of the isothermal temperature coefficient is larger in the PSCPM based model compared to the CELL based model (approximate difference is .4 pcm/F). The resulting earlier TRINODE calculations included analytically the use of a burnable absorber dependent adjustment. This analytical adjustment was used as part of the TRINODE methodology to produce the results presented in Table 3.2.1. The Salem 1 Cycle 8 CELL model results presented in Table 4.2.1 did not use the prior adjustment. The extended burnup model does not have a similar adjustment built into the methodology; instead a constant bias is applied after the TRINODE calculation (Table 3.2.1 shows the mean = 0.0 vs Table 4.2.1 shows a mean of 1.31).

7 *What is causing the Revision 1/Revision 2 bias in the isothermal temperature coefficient?*

See the response to item 6 above.

8 *Is the calculation of the delayed neutron fraction β_{eff} based on the ENDF/B-V data? If not, provide justification for the value used.*

PSE&G does not use the β values contained in the output edits of PSCPM nor the β values based on ENDF/B-V. The PSE&G methodology uses a PSE&G code called BETA to calculate core β_{eff} values and the input β values to this calculation are from Tuttle (see references at the end of this response). The justification for the delayed neutron fractions used at Salem is based on an examination of the BETA code process and also on the component information that goes into the calculation of a core β_{eff}. The following paragraphs first explain the PSE&G methodology and then discuss the component parts and the resulting impact of differences in the components on the core β_{eff}.

The core β_{eff} used in point kinetics calculations is calculated by the PSE&G BETA code. The BETA methodology is shown below:

The three dimensional core average delayed neutron fractions are calculated as:

$$\beta_m = \frac{\sum_{l=1}^{lm}\sum_{k=1}^{km}(P_{l,k})^{(PWF)} \times \sum_{n=1}^{7}(\beta_{n,m} \times \nu_n \times F_{n,l,k})}{\sum_{l=1}^{lm}\sum_{k=1}^{km}(P_{l,k})^{(PWF)} \times \sum_{n=1}^{7}(\nu_n \times F_{n,l,k})}$$

where;

m = delayed neutron group, 1 to 6
l = assembly, 1 to 193
k = axial node, 1 to 12
$P_{l,k}$ = nodal power from TRINODE for assembly l, node k
PWF = power weighting factor
n = nuclide number, 1 to 7
$\beta_{n,m}$ = delayed neutron fraction for nuclide n, group m
ν_n = average number of neutrons per fission for nuclide n
$F_{n,l,k}$ = fission fraction for nuclide n, assembly l, node k

The total delayed neutron fraction is calculated as:

$$\beta_{tot} = \sum_{m=1}^{6} \beta_m$$

The effective delayed neutron fraction is calculated using the importance factor, I, as:

$$\beta_{eff} = I \times \beta_{tot}$$

In the BETA methodology the $P_{l,k}$ comes from the TRINODE case which simulates the reactor condition of interest and the $F_{n,l,k}$ comes from the cycle specific full core PDQ case. Thus these P and F values reflect the nuclear data library used in the lattice code, PSCPM. The $B_{n,m}$ and the ν_n are input and may come from any external source. To determine the impact of using the current PSE&G β values (Tuttle) versus the ENDF/B-V β values, two BETA calculations were performed for a typical Salem 1 Cycle 9 BOC condition. The table below summarizes the β input data and the BETA code output results:

Nuclide	Current PSE&G Values	ENDF/B-V Values
U-235	0.00683	0.00685
U-238	0.01639	0.01570
Pu-239	0.00224	0.00223
Pu-241	0.00532	0.00549
BETA results		
βeff	0.00618	0.00627

As discussed in Section 3.5 of the Salem topical report, the uncertainties in the calculation of β_{eff} are composed of several components, the most important being:

a. experimental values of β and λ by nuclide

b. calculation of the spatial nuclide inventory

c. calculation of core average β_{eff} as a flux weighted average over the spatial nuclide inventory

d. calculation of β_{eff} from the core average as $\beta_{eff} = \beta$ * I , where I = importance factor.

The first three items are relevant to the justification of the β_{eff} values to be used by PSE&G. For item a), the previous table showed the β by nuclide. This data contains the uncertainties of the experimental data and of the processing of the experimental data. The major significant difference is the U-238 values. When the BETA calculations are performed in which the only difference is the values of β by nuclide, the impact of the current PSE&G values versus ENDF/B-V results in approximately 1.5% smaller β_{eff} values.

For item b), the most important nuclide concentrations with respect to β are U-238, U-235 and Pu-239. An inspection of the EPRI-CELL versus the EPRI-CPM ability to predict nuclide concentrations (Table 3.4.1 and Figures 3.4.1-3 vs Table 4.4.1 and Figures 4.4.1-3) shows that EPRI-CPM agrees better with experiment and for the three nuclides of interest the difference between measured and predicted is less than 1%.

For item c), the uncertainty associated with the calculation of a core average β is dependent on the relative flux weighting of the individual assemblies in the core. This uncertainty does not change due to the use of different values of β by nuclide. The item d) uncertainty also does not change due to the use of different values of β by nuclide.

Thus, using the same approach as pages 3-21 and 3-22 of the Salem topical report, the combined uncertainty for the extended cycle burnup model using the PSE&G BETA methodology is:

$$1.5\%(a) + 1.0\%(b) + 0.8\%(c) + 0.5\%(d) = 3.8\%$$

Based on this review of the PSE&G BETA methodology and the items that influence the calculation of the core β_{eff}, it is concluded that the 4% reliability factor for β stated in Tables 3.0.1 and 4.0.1 is appropriate for the PSE&G BETA methodology.

BETA Data Source References:

Tuttle, R.J., "Delayed - Neutron Yields in Nuclear Fission," INDC (NDS)-107/G+ Special, from "Proceedings of the Consultant's Meeting on Delayed Neutron Properties," Vienna, (March 1979).

Tuttle, R.J., "Delayed - Neutron Data for Reactor Physics Analysis," NSE-56 37-71 (1971).

9 *Provide the fuel exposure dependence of the confidence limits for X(I, K, M) and X(I, M) for the extended burnup model (Figures 3.6.15 and 3.6.18 for the Revision 1 model).*

Figure 1 shows the exposure dependence for both models. The confidence limits for both the normal and the non-parametric statistics for all sub-populations fall below the bounding value reliability factors for F_Q and $F_{\Delta H}$.

10 *What is the maximum fuel exposure for which the extended burnup will be applied? Discuss the adequacy of the benchmarking for these fuel exposures.*

The lead assembly maximum burnup to which the extended burnup model will be applied is 65,000 MWD/MTU. This burnup bounds current fuel utilization plans.

For current fuel designs used at Salem, the maximum burnup is about 55,000 MWD/MTU for the lead assembly and 52,000 for the batch average. Future cycles are being planned for which the lead assembly burnup will be approximately 60,000 and 55,000 for the batch average. As shown in the table in the background discussion, the lead assembly discharge burnup has gradually increased with time. The on-going model validation process ensures that the model adequately predicts the behavior of future reloads. This is illustrated in Attachment 1 of this document, which shows the use of a scatter plot for the difference between measured and predicted integral power as a function of assembly burnup. Scatter plots such as this are used to determine if model biases or trends exist. In addition, the reliability factors are updated if necessary based on the on-going benchmarks. If the on-going benchmarking shows that smaller reliability factors are warranted, then the Salem Reactor Physics topical report will be revised.

11 *How has the change from EPRI-CELL/PDQ to PSCPM affected the determination of the detector fission rate-to-assembly power ratios? Is this change included in the extended burnup model benchmark comparisons?*

There has been no change in the method for determining this ratio. Both the CELL based (Figure 2.0.1) and the PSCPM based (Figure 2.0.2) methods utilize PDQ for generation of the instrument factors (detector fission rate/assembly power). The box labeled "Normalization Reaction Rates Pin to Box" in both figures is the step which generates the instrument factors to be fed to the SIGMA code which performs the measured vs predicted reaction rate comparisons. Any change in the numerical values between the models is derived from PDQ's cross sections coming from EPRI-CELL or from PSCPM.

The extended burnup model benchmark comparisons include the use of the PSCPM/PDQ instrument factors.

12 *Provide all references used in determination of the Doppler bias and reliability factors.*

1 J. Fisher, R. Grow, D. Hodges, J. Rapp, K. Smolinske, Utility Resource Associates. "Evaluation of Discrepancies in Assembly Cross Section Generator Codes", Volumes 1 & 2, 1989, Volume 3, 1990. EPRI-NP-6147.

2 J. Fisher, R. Grow, Utility Resource Associates, "Evaluation of Discrepancies in Assembly Cross Section Generator Codes", Volume 4 Doppler Evaluations,July 1993, EPRI NP-6147.

3 "Power Coefficient Validation of the Kewaunee ARMP Model." ANS Paper Circa 1981.

4 E. Hellstrand, P. Blomberg, S. Horner, "The Temperature Coefficient of the Resonance Integral for Uranium Metal and Oxide." Nuclear Science & Engineering, July 12, 1990, 8-497-506.

5 D. W. Dean. "Analysis of Axial Xenon Oscillations at Prairie Island." International Reactor Physics Conference, Jackson Hole, WY. September 18-22, 1988.

6 "Results of PI2 Cycle 13 Xenon Test." March 19, 1991. NSP Internal Report, K. L. Wright

7 R. D. Mosteller, L. Eisenhart, S. Levy Inc. "ENDF/B-V Doppler Evaluations." Presented at Eight Semi Annual EPRI Reactor Physics Users Group. November 1989.

8 R. D. Mosteller, L. Eisenhart, S. Levy Inc.; R. C. Little, Los Alamos National Lab; W. J. Eich, J. Chao, Electrical Power Research Institute. "Benchmark Calculations for Doppler Coefficient of Reactivity."

Table 1

Response to Question 3, List of TRINODE Changes

1 Table look up of nodal K^{∞} and $\Delta\rho$ values for Doppler, boron, xenon, and control rods; M^2, σ_{xe}, Σ_F, and υ/κ as functions of number of burnable poison fuel pins, number of BPR rodlets, exposure, moderator temperature history, instantaneous moderator temperature, fuel temperature, and boron concentration.

2 Add moderator temperature history input, calculation, and edits.

3 Add target K_{eff} and convergence criterion to boron search logic.

4 Change the definition of the control rod search target band from ΔI to axial offset.

5 Add a $\Delta\rho_{poison}$ core map edit for burnable poison fuel pins.

6 Calculate the Doppler fuel temperature for table look up as a function of nodal power.

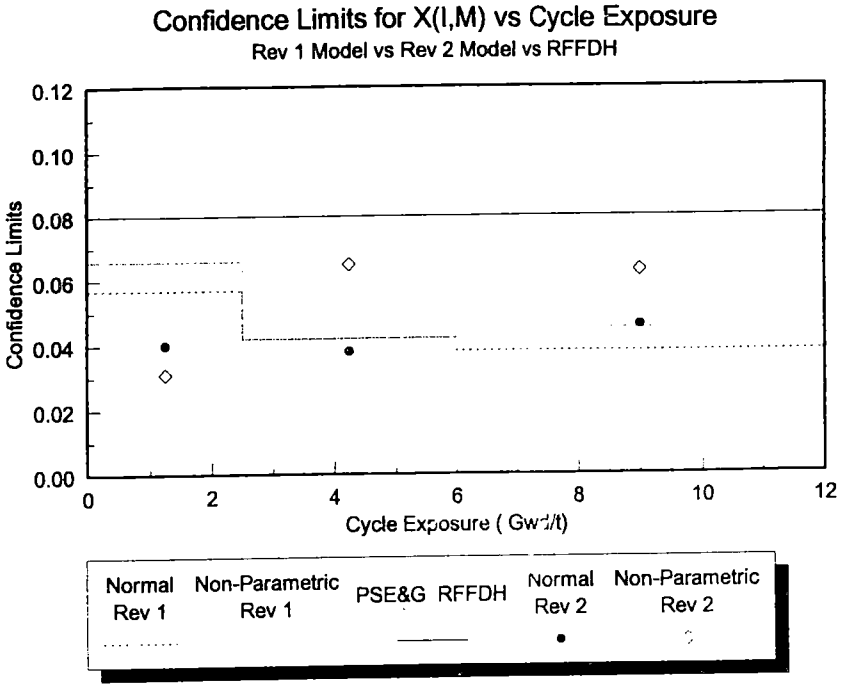
7 Allow the axial variations in assembly designs to be modeled by assigning fuel types by axial level in each assembly.

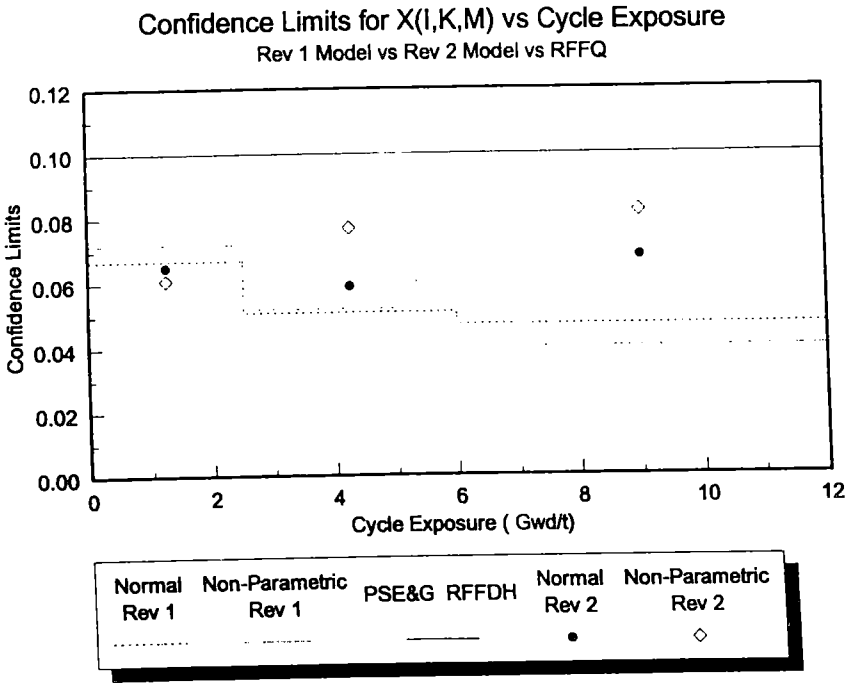
8 Allow optional fuel temperature, TFUEL, input. This will set all nodal fuel temperatures equal to TFUEL.

9 Model the finite lattice effect on control rod worth. The finite lattice effect accounts for the reduction in rod worth in a core geometry compared to the PSCPM single assembly rod worth in an infinite lattice geometry.

Figure 1

Response to Question 9





SUMMARY AND CONCLUSIONS

This document provides PSE&G's response to the NRC request for additional information for the PSE&G Salem Reactor Physics Methods Topical Report, NFU-0039, Revision 2. As the result of our on-going model benchmarking and validation process additional benchmarking work has been completed (Attachment 1) since the topical report submittal. Based on this work and the recent Doppler work described in response #1 which was sponsored by EPRI, PSE&G will utilize the extended burnup model with the biases and reliability factors shown in the table below. Note that the temperature coefficient and Doppler defect biases have been revised since NFU-0039, Revision 2 was submitted. These changes are based on both the new PSE&G benchmarking and the EPRI results.

Revised Reliability Factors and Biases for PSE&G Extended Burnup Model Applied to Salem		
Parameter	Reliability Factor	Bias
Rod Worth		
Meas ≥ 600 pcm	RFROD = 15%	0
Meas < 600 pcm	RFROD = 100 pcm	0
Totals	RFROD = 10%	0
Temperature Coefficient		
Moderator (MTC)	RFMTC = 2.1 pcm/°F	1.3
Isothermal (ITC)	RFITC = 2.1 pcm/°F	1.7
Doppler		
CAE ≤ 32	RFDC = 10%	-(25% - .3125 *CAE)
CAE > 32	RFDC = 10%	-15%
Doppler Defect		
CAE ≤ 32	RFDD = 10%	-(25% - .3125 *CAE)
CAE > 32	RFDD = 10%	-15%
Power Distribution		
F_Q		
P ≥ .50	RFFQ = 0.10	0
P < .50	RFFQ = 0.16 - (0.12 * P)	0
$F_{\Delta H}$		
P ≥ .30	RFFDH = 0.08	0
P < .30	RFFDH = 0.09 - (0.033 * P)	0

where: CAE=Core Average Exposure in Gwd/t,and P= Core rated thermal power

Attachment 1

Most Recent Benchmark Results

Since the submittal of Revision 2 of NFU-0039, Salem Reactor Physics Methods, three additional cycles have completed operation at Salem. As shown in the BACKGROUND section discussion, these cycles utilized the multiple burnable poison design. Comparisons of measured versus predicted rod worths, temperature coefficients and reaction rates are made as part of the on-going model validation. In response to question 4 the data from these recent cycles has been compiled in summary format in Tables A.1 and A.2. Table A.1 lists the three recent unit/cycles and also the flux maps analyzed. These three most recent cycles are compared to the results for eight cycles (the initial five plus the recent three) in Table A.2.

Rod Worth Benchmarking

The rod worth results (mean and standard deviation) are shown in Table A.2 for each of the three groups of cycles for the categories of measured worth equal to or greater than 600 pcm, measured less than 600 pcm and the total worth of all rods. The eight cycle results in Table A.2 can be compared to the extended burnup model reliability factors (Table 4.0.1 topical report) of RFROD = 15% for individual rods Meas $\geq$ 600, RFROD = 100 pcm for individual rods Meas < 600 pcm, and RFROD = 10% for total rod worth. Table A.2 shows that the results are consistent between the different groupings of cycles.

Isothermal Temperature Coefficient Benchmarking

Only one ITC measurement per cycle was made for the recent cycles in contrast to the many measurements for the earlier Salem cycles. For the extended burnup model the mean difference (M - P) for the eight cycles is +1.7 pcm/F. A meaningful reliability factor can not be determined for the recent cycles alone due to the small sample size, thus the alternate approach used for small populations was to show on a bounding value approach that all new data points lie inside the ITC bias plus reliability factor. Based on these results the ITC bias was changed to 1.7 pcm/F, the MTC bias to 1.3 pcm/F (accounts for subtracting out the Doppler contribution to the bias, see response to #6), and the reliability factors remained at 2.1 pcm/F for both the ITC and MTC.

Power Distribution Benchmarking

Table A.2 shows the integral and local power distribution results. The results in Table A.2 can be compared to the extended burnup model reliability factors (Table 4.0.1 topical report) of RFFDH = 0.08 for power ≥ 30% and RFFDH = 0.09 - P/30 for power < 30% ; and RFFQ = 0.10 for power ≥ 50% and RFFQ = 0.16 - (0.12*P) for power < 50% . Table A.2 shows that the results are reasonably consistent between the different groupings of cycles.

Figure A.1 shows the scatter plot of the M-C integral residual for each assembly monitored versus assembly burnup at the time of the measurement. This data shows consistent model performance with increasing burnup.

Table A.1 - Recent Cycle Flux Map Database

Salem 2 Cycle 7			
Unit/Cycle/Map	Cycle Burnup	% Power	D Bank Position
2-701	17	21.8	173.00
2-702	66	46.3	186.00
2-705	2152	99.1	228.00
2-708	5134	99.2	228.00
2-712	9324	99.8	228.00
Salem 1 Cycle 10			
1-1001	8	23.8	177.00
1-1002	18	46.5	180.00
1-1004	1269	100.0	228.00
1-1010	7863	99.9	228.00
1-1013	9955	100.2	228.00
Salem 1 Cycle 11			
1-1101	8	20.1	156.00
1-1102	47	46.0	177.00
1-1105	2033	99.9	228.00
1-1108	4652	99.8	228.00
1-1112	8109	81.3	186.00
1-1116	11692.00	99.9	228.00

Table A.2
Summary of Statistical Results

	Recent 3 Cycles	All 8 Cycles	8 Cycle Reliability Factors
1) Rod Worths (μ,σ)			
$M \geq 600$ (M-C)/C	2.6% ± 5.6%	2.8% ± 3.9%	8.6%
$M < 600$ (M-C)	6.1pcm ± 44.6 pcm	4.2pcm ± 29.9 pcm	66 pcm
Total Worth (M-C)/C	1.9% ± 1.8%	2.1% ± 1.7%	5.7%
3) $F_{\Delta H}$ (σ) (M-C)			
High Power Maps	0.025	0.023	.040
Mid Power Maps	0.030	0.026	.046
Low Power Maps	0.036	0.029	.052
4) F_Q (σ) (M-C)			
High Power Maps	0.041	0.045	.075
Mid Power Maps	0.063	0.062	.104
Low Power Maps	0.065	0.077	.129

Figure A.1
Integral Power Distribution Accuracy vs Assembly Burnup