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CALCULATION SUMMARY SHEET (CSS)

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Title REACTOR RECORD UNCERTAINTY DETERMINATION

PREPARED BY:		REVIEWED BY:	
		METHOD: ☒ DETAILED CHECK ☐ INDEPENDENT CALCULATION	
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TITLE PRIN. ENG.	DATE ~~12/08/04~~ 12/6/04	TITLE ENGINEER	DATE ~~12/08/04~~ 12/06/04 MS
COST CENTER 212020	REF. PAGE(S) 21	TM STATEMENT: REVIEWER INDEPENDENCE	WJA

PURPOSE AND SUMMARY OF RESULTS:

This calculation evaluates burnup uncertainty for PWR and BWR fuel assemblies for using data from utility records. The records give the difference between calculated and measured burnup for fuel assemblies at the end of each fuel cycle. Evaluated here are the uncertainties in calculated minus measured burnup (*D*) and he percent difference of calculated to measured burnup (*P*), given by the equation:

$$P = 100 \text{ (calculated burnup – measured burnup) / (measured burnup)}$$

The assembly data (Ref. 5) include nine PWR plants of two designs, designated as A and B. The six A plants are identified as A1, A2, A3, A4, A5, and A6. The three B plants are identified as B1, B2, and B3. No comparable burnup data was found for BWRs. Thus, BWR burnup uncertainty is evaluated by considering the BWR radial power uncertainty factors (CNRFs) from Ref. 5 in combination with the PWR burnup uncertainty results.

This evaluation uses a representative sample of industry data consisting of 5.447 assemblies having end-of-cycle (EOC) burnup >10,000 MWd/MTU. More than 10 cycles of data were available for A1, A2, B1, B2, and B3. Only one recent cycle of data was available for each A3, A4, A5, and A6. Due to the smaller sample size and recent timing of the records for these plants, these four cycles are combined in a separate evaluation from the other five plants.

The data for *D* and *P* for each plant are grouped by burnup and tested for normality. The groupings are: burnup > 10,000, burnup of 10,000-30,000, burnup >30,000, identified as Bin 1, Bin 2, and Bin 3, respectively. If normal, the uncertainty for *D* and *P* is determined with a common one-sided tolerance limit method at 95% confidence and 95% probability. Further normality testing determines whether groups of plants can be combined for further burnup uncertainty evaluation. For individual plants, the uncertainty values for *P* range from 2.4% to 3.8% in Bin 1, from 2.7% to 4.2% in Bin 2, and from 2.0% to 3.2% in Bin 3. For each plant, the *P* uncertainty is less than the CNRF, which has a range on the order of 3-5% for the A and B plants. In the analysis for A3, A4, A5, and A6, combined, the uncertainty values for *P* are 1.0%, 1.3%, and 1.0% for Bins 1-3, respectively. Normality testing shows some plants can be grouped. Plants A1 and A2 can be combined to obtain an uncertainty of 2.7% for *P* for Bin 2. Plants B1, B2, and B3 can be combined for *P* in all three bins to give an uncertainty of 3.1% in Bin 1, 3.6% in Bin 2, and 2.3% in Bin 3. A1 and A2 can not be grouped for Bins 1 and 3. The combined data for A1, A2, B1, B2, and B3 does not pass the normality testing.

The calculated *P* uncertainty values are less than the CNRF in Ref. 5 for each plant evaluated. The CNRF range is on the order of 3-5% with higher values in effect for earlier fuel cycles. The results of this evaluation for *P* uncertainty indicate that the CNRF for each reactor provides a conservative estimate of burnup uncertainty, as expected. Moreover, since the BWR CNRFs fall within the same range as those for the PWRs, it is expected that the *P* uncertainty obtained from an analysis of BWR data would give values no greater than the largest value for *P* (4.2%) in the PWR uncertainty analysis.

The burnup uncertainties will be used to adjust either the waste package loading curves or the burnup values of assemblies shipped to the repository.

The engineering calculation supports the burnup credit methodology in Ref. 1 and is performed in accordance with the AREVA/FANP procedures (Ref. 2 and Ref. 3). Revision 03 of this calculation does not affect its results in any way. Changes made were to clarify references and traceability of the data.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:		THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK
CODE/VERSION/REV	CODE/VERSION/REV	
		☐ YES ☒ NO

Revision Number	Date
02	May 2004

- The footnote on Tables 3A, 4A, 5A, 6A, 7A, 8A and 9A is changed to "Kσ is the uncertainty in calculated - measured burnup in MWD/MTU".
- The period is removed at the end of the title of Sections 5.2.8 and 5.2.9.
- The following sentence is added to the first paragraph of Section 1 and at the end of Section 6: "The burnup uncertainties will be used to adjust either the waste package loading curves or the burnup values of assemblies shipped to the repository."

03	December 2004

- Amended Calculation Summary Sheet (CSS) by adding statement that no results in this calculation were affected by this revision.
- Inserted referral to Reference 4, page 7 of 129.
- Updated Reference 2 and 3, version and date, page 21 of 129.
- Added Reference 4 'Data Verification Letter' to listing of references, page 21 of 129. Reference 4 was previously not used in Reference Section.
- Completed Design Verification Checklist to reflect revisions.

1. PURPOSE

The objective of this calculation is to evaluate commercial spent nuclear fuel (CSNF) burnup uncertainty based on pressurized water reactor (PWR) and boiling water reactor (BWR) records kept by each utility. The burnup uncertainties will be used to adjust either the waste package loading curves or the burnup values of assemblies shipped to the repository.

This engineering calculation supports the burnup credit methodology in Reference 1 and is performed in accordance with the AREVA/FANP procedures in References 2 and 3.

2. METHOD

Values of the radial power distribution and burnup for commercial nuclear fuel assemblies are determined from calibrated calculations that are continually verified with in-core measurements throughout the in-core irradiation history of the fuel assemblies. These values are documented in proprietary core operations records kept by each utility that operates commercial PWR and BWR reactors (Reference 4).

Burnup measures the exposure of nuclear fuel during reactor core power production and is usually expressed in units of GWd or MWd per MTU initially loaded into a fresh assembly. For each cycle of reactor operations, core operations reports provide measured burnup determined from an array of calibrated in-core detectors and calculated burnup determined from calculational models of the reactor core power distribution. The measured and calculated burnups are used to determine the difference between calculated minus measured burnup (D) and the percent difference of calculated to measured burnup (P) as follows:

$$D = \text{calculated burnup} - \text{measured burnup}$$

$$P = 100\ (\text{calculated} - \text{measured}) / \text{measured}$$

Reactor records give the calculated burnup and measured burnups from which D and P are calculated. This calculation determines burnup uncertainty values based on a statistical analysis of D and P for nine PWR reactors for which data is documented in Reference 5.

Six reactors in this calculation have a similar plant design for fuel assemblies and in-core detectors and are designated as plants A1, A2, A3, A4, A5, and A6. The other three plants have a similar plant design for fuel assemblies and in-core detectors and are designated as plants B1, B2, and B3. The data used in this analysis are from fuel assemblies having either a 15 x 15 or a 17 x 17 array of fuel rods.

A search was done and no data was found for pairs of measured and calculated burnup data for BWR fuel assemblies. A proprietary report was obtained for BWR reactors that gives calculational nuclear reliability factors (CNRF), which express the uncertainty in the calculated radial power versus measured radial power in BWR reactors. These BWR CNRFs are listed in Reference 5. Since radial power uncertainties are directly related to burnup uncertainties, the

7. REFERENCES

1. AREVA/FANP Document Number 38-5032055-01, 2003. YMP. *Disposal Criticality Analysis Methodology Topical Report*, YMP/TR-004Q, Rev. 02. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. DOC.20031110.0005.

2. AREVA/FANP Administrative Procedure, Number: 0402-01, Preparing and Processing FANP Calculations, November 2003, Framatome ANP, Lynchburg, VA 24506

3. AREVA/FANP Document Number FQM Rev 01, 2003. Framatome ANP, Inc. Fuel Sector Quality Management Manual (US Version)

4. AREVA/FANP Document Number 38-2200201-00, December, 2004. *Data Verification Letter NFP-04-111-A, 16 November, 2004.*

5. AREVA/FANP Document Number 32-2200138-00, 2004. Reactor Record Evaluations for Burnup Uncertainty.

6. AREVA/FANP Document Number 32-5037984-00, 2003. Critical Limit Development for PWR and BWR SNF Waste Package.

7. D'Agostino, R.B. and Stephens, M.A. 1986. *Goodness-of-fit Techniques*. Statistics, Textbooks and Monographs, Volume 68. New York, New York: Marcel Dekker.

8. Natrella, M.G. 1963. *Experimental Statistics*. National Bureau of Standards Handbook 91. U.S. Government Printing Office, Washington, D.C.

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DESIGN VERIFICATION CHECKLIST

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Title Reactor Record Uncertainty Determination

1.	Were the inputs correctly selected and incorporated into design or analysis?	☐ Y	☐ N	☒ N/A
2.	Are assumptions necessary to perform the design or analysis activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent re-verifications when the detailed design activities are completed?	☐ Y	☐ N	☒ N/A
3.	Are the appropriate quality and quality assurance requirements specified? Or, for documents prepared per FANP procedures, have the procedural requirements been met?	☒ Y	☐ N	☐ N/A
4.	If the design or analysis cites or is required to cite requirements or criteria based upon applicable codes, standards, specific regulatory requirements, including issue and addenda, are these properly identified, and are the requirements/criteria for design or analysis met?	☒ Y	☐ N	☐ N/A
5.	Have applicable construction and operating experience been considered?	☐ Y	☐ N	☒ N/A
6.	Have the design interface requirements been satisfied?	☐ Y	☐ N	☒ N/A
7.	Was an appropriate design or analytical method used?	☐ Y	☐ N	☒ N/A
8.	Is the output reasonable compared to inputs?	☐ Y	☐ N	☒ N/A
9.	Are the specified parts, equipment and processes suitable for the required application?	☐ Y	☐ N	☒ N/A
10.	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?	☐ Y	☐ N	☒ N/A
11.	Have adequate maintenance features and requirements been specified?	☐ Y	☐ N	☒ N/A
12.	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?	☐ Y	☐ N	☒ N/A
13.	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?	☐ Y	☐ N	☒ N/A
14.	Has the design properly considered radiation exposure to the public and plant personnel?	☐ Y	☐ N	☒ N/A
15.	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?	☐ Y	☐ N	☒ N/A
16.	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?	☐ Y	☐ N	☒ N/A
17.	Are adequate handling, storage, cleaning and shipping requirements specified?	☐ Y	☐ N	☒ N/A
18.	Are adequate identification requirements specified?	☐ Y	☐ N	☒ N/A
19.	Is the document prepared and being released under the FANP Quality Assurance Program? If not, are requirements for record preparation review, approval, retention, etc., adequately specified?	☒ Y	☐ N	☐ N/A

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Comments:

See Record of Revisions page for changes to References. No other parts were affected.

Verified By:	Mehmet Saglam	M. Saglam	12/6/2004
(First, MI, Last)	Printed / Typed Name	Signature	Date