TRANSNUCLEAR
AN AREVA COMPANY

June 22, 2009
E-28237

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Revision 1 to Transnuclear, Inc. (TN) Application for Revision 3 to Certificate of Compliance No. 9302 for the Model No. NUHOMS®-MP197 Packaging – Response to Requests for Supplemental Information (Docket No. 71-9302, TAC No. L24336)

Reference: Letter from Eric Benner (NRC) to Donald Shaw (TN), "APPLICATION FOR REVISION TO CERTIFICATE OF COMPLIANCE NO. 9302 FOR THE MODEL NO. NUHOMS®-MP197 PACKAGING, DOCKET NO. 71-9302 – SUPPLEMENTAL INFORMATION NEEDED," June 8, 2009

The letter referenced above advised TN that NRC staff had completed an acceptance review of our April 14, 2009 application for revision to Certificate of Compliance No. 9302 for the Model No. NUHOMS®-MP197 Packaging and that supplemental information is needed for the staff to continue their review. The information needed was enclosed in the letter as Requests for Supplemental Information (RSIs). The letter also included observations to allow TN to start earlier on items containing the potential to be asked at a later date. The letter indicated that responses to the observations are not required for the staff to begin a detailed technical review.

The purpose of this submittal is to respond to the RSIs and most of the observations. Enclosure 2 provides the RSIs and observations, and TN responses. Enclosure 3 provides a list of replacement, new, or removed NUHOMS®-MP197 proprietary Safety Analysis Report (SAR) pages, and the reasons for changes. These pages are annotated Rev. 6, 06/09. Changed areas are indicated by revision bars and italics for inserted text. Enclosure 4 provides the replacement and new SAR pages. Enclosure 5 provides a list of replacement, new, or removed non-proprietary SAR pages. Enclosure 6 provides the replacement and new pages.

A small number of changes to table cross-referencing, misspellings, etc. not related to the RSIs or observations are also included in the SAR changed pages. These are explained in the Enclosure 3 reasons for change.

Enclosure 4 of this submittal includes proprietary SAR information which may not be used for any purpose other than to support staff review of the application. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure.

Enclosure 7 is a technical document, mentioned in the response to Criticality Observation No. 3, which is a new SAR reference and is provided herein to support staff review of the application.

7135 Minstrel Way, Suite 300, Columbia, MD 21045
Phone: 410-910-6900 • Fax: 410-910-6902

NMSS01

Should the NRC staff require additional information to support review of this application, please do not hesitate to contact Mr. Don Shaw at 410-910-6878 or me at 410-910-6881.

Sincerely,

Jayant Bondre, Ph.D.
Vice President - Engineering

cc: Christopher Staab (NRC SFST) (six copies of this cover letter and Enclosures 1 through 4, and 7, provided separately)

Enclosures:

1. Affidavit Pursuant to 10 CFR 2.390
2. RSIs, Observations, and Responses
3. List of Changed NUHOMS®-MP197 Safety Analysis Report Pages, Revision 6 (for the Proprietary version)
4. Replacement and New NUHOMS®-MP197 Safety Analysis Report Pages, Revision 6 (for the Proprietary version)
5. List of Changed NUHOMS®-MP197 Safety Analysis Report Pages, Revision 6 (for the Non-proprietary version)
6. Replacement and New NUHOMS®-MP197 Safety Analysis Report Pages, Revision 6 (for the Non-proprietary version)
7. B. ROQUE, A. Santamarina, "Experimental Validation of Actinide and Fission Products Inventory from Chemical Assays in French PWR Spent Fuels," (cover sheet and Pages 69 to 82 of IAEA-TECDOC-1378, "Practices and developments in spent fuel burnup credit applications," Proceedings of an Technical Committee meeting, Madrid, 22-26 April 2002, published October 2003)

AFFIDAVIT PURSUANT TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Jayant Bondre, depose and say that I am Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosure 4 and as listed below:

1) SAR Appendix A.2.13.11
2) Portions of SAR Chapter A.6
3) Portions of SAR Appendix A.6.5.4
4) Portions of SAR Appendix A.6.5.5
5) Portions of SAR Appendix A.6.5.6
6) Portions of SAR Appendix A.6.5.7

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, Inc. in designating information as a proprietary trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced documents, should be withheld.

1) The information sought to be withheld from public disclosure involves certain safety analysis report analyses related to the design of the modified NUHOMS®-MP197 transport cask which are owned and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.390 with the understanding that it is to be received in confidence by the Commission.

4) The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

5) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. and to other owners of the information because:

a) A similar product is manufactured and sold by competitors of Transnuclear, Inc.

b) Development of this information by Transnuclear, Inc. and other owners of the information required expenditure of considerable resources. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

c) In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of a design and analysis of a dry spent fuel transportation system.

d) The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e) The information consists of certain safety analysis report analyses related to the design and analysis of dry spent fuel storage and transportation systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to unfairly get a better competitive position with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, while avoiding the expense of developing similar data and analyses in support of their processes, methods or apparatus.

f) In pricing Transnuclear, Inc.'s products and services, significant research, development, engineering, analytical, licensing, quality assurance and other costs and expenses must be included. The ability of Transnuclear, Inc.'s competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

Further the deponent sayeth not.

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 22nd day of June, 2009.

Notary Public

My Commission Expires 10/14/2012

Structural

RSI-1: Provide pertinent and adequate justification, (including the applicability of the results of the Fuel Integrity Project (FIP) Program) for use of a fracture mechanics approach to demonstrate the adequacy of high burn-up spent fuel cladding during transportation.

The applicant has been made aware of staff's position in the past, on previous applications (TN-68, Standardized NUHOMS Amendment 10) regarding use of fracture mechanics approach to establish adequacy of high burn-up spent fuel cladding. The case presented in Addendum 10 f, g, and h based on French Test (FIP Program) data, does not provide a justifiable basis and is relying on fuel for burn-up to only approximately 50 GWd/MTU. The linear interpolation from 50 to 65 GWd/MTU is not substantiated, and therefore not acceptable.

This information is required by the staff to verify the compliance with 10 CFR 71.71 and 10 CFR 71.73 regulations.

TN Response:

The following is a summary of fuel cladding allowable fracture toughness (K_{IC}) approaches used in TN applications:

- *TN-68: TN developed a correlation model to calculate the fracture toughness for the high burnup fuel based on the values from the low burnup fuel.*
- *CoC 1004 Amendment 10: TN used the lowest value of K_{IC} from EPRI report "Fracture Toughness Data for Zirconium Alloys"; the value used is 16.36 ksi in$^{1/2}$.*
- *CoC 9302 Revision 3 Application (for the MP197HB): TN developed the K_{IC} values by using the TNI proprietary FIP fuel rod samples tests with burnups between 49.6 and 49.8 GWd/tU. The KIC values are estimated for high burnup by comparing the areas under the stress strain curves with similar stress strain curves for low burnup tests.*

Note that the approach has been different in all applications to calculate the fracture toughness values of high burn-up fuel. All the fracture mechanics evaluations TN had performed and documented in the MP197HB submittal were intended to show that the fuel "can be retrieved and handled by normal means" even after the transportation normal or accident conditions, which is not necessary or required under 10CFR Part 71. Therefore, all the brittle fracture evaluation of the damaged fuel is removed from the submittal.

As pointed out by the Staff in the Acceptance Review Letter, according to 10CFR71.55(d) under normal conditions of transport:

(1) the contents would remain subcritical, and
(2) the geometry form of the package contents would not be substantial altered

Also according to 10CFR71.55(d):

(1) the fissile material must be assumed to be in its most reactive credible configuration consistent with the damaged condition

TN has demonstrated in Appendix A.6 that the criticality analysis of the damaged fuel assumes the most credible configuration of damaged fuel assemblies consistent with the

damaged fuel condition of the package and chemical and physical form of the contents and they don't need to maintain their integrity. The results of the criticality analysis show that the package remains subcritical under all credible scenarios. All the damaged fuel assemblies are stored in a fuel compartment that includes both the top and bottom end caps. Therefore the damaged fuel assemblies are captured in the basket fuel compartment by end caps for maintaining the critical configuration.

The definition of the damaged fuel is also revised to be consistent with the criticality analysis. Therefore, the existing damaged fuel criticality analysis documented in Appendix A.6 meets all the requirement of 10CFR71.

Observation-1: The applicant should consider providing a table listing nine different Dry Shielded Canisters (DSCs) that go into the Package MP-197 HB, with information such as major physical dimensions, contents, heat loads, low-high burn up, etc.... This may provide for a more efficient review by the staff.

TN Response:

The information requested above is already present in the submitted SAR application and can be found at the following locations in the SAR. The DSC dimensional information is summarized in Chapter A.3, Table A.3-1. The maximum heat loads allowed are also provided in Chapter A.3, Table A.3-8 for BWR fuel and Table A.3-9 for PWR fuel. Detailed information concerning the fuel assemblies contained in each DSC type is provided in the specific Appendix for each DSC that follows Chapter A.1. Burnup limits are provided in Tables for each DSC that also are contained in the Chapter A.1 Appendices. As an example, for the 24PTH DSC, Tables A.1.4.3-5 and A.1.4.3-8 in Appendix A.1.4.3 contain the burnup, enrichment, and cooling time requirements. Note that the allowable burnup values vary with decay time, and therefore each table has a large number of allowable burnup values which does not lend itself to a single summary table.

Observation-2: The applicant has not identified deviations from staff's guidance documents. The applicant should provide explanations why the latest revisions of computer codes ANSYS and LS-DYNA (for performing structural analysis of regulatory drop scenarios) were not used, as discussed in ISG-21. Also, the applicant should provide discussions of implications, if any, of not using the latest revisions of these computer codes on the analyses results.

TN Response:

The versions of ANSYS and LS-DYNA computer codes that were used during the MP197HB analyses reflect the time when the analyses were initiated. Several of these efforts date back over more than one release of the computer code. Rather than revise analyses because of code updates, the original version of the code was maintained through the design process. Note that all versions of the codes used were maintained via updates and error messages from the developer of the software consistent with Transnuclear Quality Procedures. In addition, any change in computer platform resulted in an update of the V&V documentation. As a result, all verification and validation (V&V) requirements continued to be met throughout the design process. Given that the code was valid at the time of release, the analyses using that code are also valid as long as the code is maintained. Note that ISG-21 does not require use of the latest release of a computer code. What is required is that ". . . adequate validation of that CMS (computational modeling software) must be demonstrated by the applicant." Thus the previous versions of the properly V&Ved software may be used even after newer versions of the same software have been released.

Materials

RSI-1: The applicant must address the deficiencies in the data base stated below, or treat all fuel with any cladding penetrations as damaged fuel and place in a damaged fuel can.

Criteria: Acceptance Evaluation of Addendum 10 f, g, h Dealing with Cladding Behavior. The applicant has provided insufficient justification that the conclusions drawn from the provided data is applicable to the conditions or contents of transportation

The three addendums deal with the behavior of the cladding. Addendum 10g "Fuel Integrity Project, Bend tests on as-irradiated fuel series 11, and 12" gave details of the sample characteristics, test apparatus to do 3 point bend tests, testing parameters and conditions, and results of the testing in the form of stress, displacement curves, and photos of the samples. Samples were fueled ~50 GWd/MTU Zircaloy-4, and -2 clad fuel rod segments. Tests were conducted at constant temperatures of 25C and 500C, and constant pressure.

The value of these tests appears small as there are no hydrogen levels stated, no decreasing stress to simulate potential hydride reorientation, and no pre-, or post- test metallography to indicate the hydride structure. Testing was only on Zircaloy cladding.

Addendum 10h provided many unidentified tables, and load deflection curves for the tests. There were also many plots of graphic deflection with no indication of how the curves were generated. The computer simulations are not an issue for the materials review.

Addendum 10f presented the analysis of the data in the other two addendums in an attempt to develop fracture toughness of high burnup cladding. Results at 50 GWd/MTU from the current tests were compared to fracture toughness measurements at lower burnups obtained from the literature. A linear extrapolation, based on the strain energy density concept that the staff does not except, was made to higher burnups. Major questions are: 1) the validity of a linear extrapolation, 2) data comparisons with no knowledge of the hydrogen levels of the current tests, and 3) applicability of results with no simulated drying affects.

This data is presented as proprietary but touches on a generic issue of high burnup fracture toughness. In order to approve this data for support of the fuel behavior, extensive review will be required of the addendums, references made in the addendums that the applicant would have to supply. Due to the far ranging implications of these addendums, and the conclusions drawn from the addendums, this issue would have been better treated in a topical report.

Use of this data for rods that are already breached (any size breach) when loaded requires answers to points 1 and 2 above. All three points must be resolved before the data can be applied to rods that may breach after the cask is dried.

The staff does not consider this an acceptable approach as presented. In past applications the information supplied to justify the fracture toughness has been rejected. In TN-68, the applicant was told that the staff does not accept the methodology and they should remove it from the application since it was not necessary. In TN Amendment 10, the margin between the calculated fracture toughness, and the measured fracture toughness was so great that the applicability of the data was not questioned further. See also Structural RSI-1.

This information is required by the staff to verify the compliance with 10 CFR 71.71 and 10 CFR 71.73 regulations.

TN Response:

See response to RSI-1 in the Structural area.

Observation-1: The applicant should revise to address the following: 1) the current definitions must be made consistent with the requirements of Part 71 and the analysis must be provided. 2) The definition can be left as is, if the applicant shows that any assembly that is retrievable will remain in a configuration that is subcritical.

(Chapter 1.A all subsections, Definition of damaged fuel.)

The applicant has not revised the definition of damaged fuel, especially the assembly, to make it relevant to transportation. Almost all definitions of damaged fuel, submitted by the applicant, contain a statement that "The extent of the damage is to be limited such that a fuel assembly needs to be handled by normal means" or "...is able to be handled by normal means and retrievability is assured following normal and off-normal event". While logically desirable, there is no Part 71 requirements for the retrievability of the spent fuel from the cask. This is a Part 72 requirement. 10 CFR 71.55(d) does require that under normal conditions of transport, 1) the contents would remain subcritical, and 2) the geometric form of the package contents (i.e. the fuel and assembly) would not be substantially altered. Also 10 CFR 71.55(e)(1) requires that the fissile material must be assumed to be in its most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents. In other words the definition for transportation should not say that the assembly will remain retrievable but that for an assembly to be considered undamaged it must not contain flaws that will allow it to entertain a critical configuration.

The staff does not consider this an acceptable approach as presented. Without acceptable definitions of damaged fuel, the staff is unable to determine the physical condition of the assemblies and their subsequent behavior under accident conditions and inspectors will be unable to determine whether the proper fuel was placed in damaged fuel can.

TN Response:

The definition of damaged fuel is revised for the 24PTH DSC in Table A.1.4.3-2, 32PTH DSC in Table A.1.4.4-2, 32PTH1 DSC in Table A.1.4.5-2, 37PTH DSC in Table A.1.4.6-2, 61BT DSC in Table A.1.4.7-1, 61BTH DSC in Table A.1.4.8-2, and 69BTH DSC in Table A.1.4.9-1 to be consistent with the criticality analysis per Response to Structural RSI-1 above.

Observation-2: The applicant should revise the Safety Analysis Report (SAR) and reference in the Certificate of Compliance (CoC), to add a plan to the Acceptance Test and Maintenance chapter to ensure that for any DSC that has spent an extended time in storage, that the contents and DSC itself meet all the conditions in the CoC. This plan should include inspections to obtain data or analysis to support the: 1) mechanical and thermal properties of the components of the DSCs related to safety, and 2) contents have not degraded during the storage period. This is needed to ensure the package will be verified to be in unimpaired physical condition after long term storage, in accordance with 10 CFR 71.87(b).

No evidence was presented to indicate that the thermal and mechanical properties of the DSCs,

or contents have not degraded during storage and are still applicable to the transportation evaluation. All the mechanical and thermal properties of the materials of construction of the DSC used in this Part 71 request are for pristine materials. Many of the DSCs were constructed and loaded many years ago, and have been on the storage pad for a considerable number of years.

The staff does not consider this an acceptable approach as presented. The materials properties used for the evaluation of the safety systems and contents of the DSCs that have already been in storage service must be representative of the conditions at the time of transport, not at the time of the loading of the DSC.

Observation-3: The applicant should provide analysis that the acceptance and qualification plans for the neutron absorber in the DSCs are consistent with the current criteria designated in Standardized NUHOMS Amendment 10.

The guidance on the qualification and acceptance of the neutron absorber material has been evolving. The current staff guidance has been adopted in TN Amendment 10. Older qualification and acceptance criteria may not be as rigorous as the currently accepted guidance used for transport. See SAR Section A.8.1.7 where additional guidance on qualification and acceptance for the neutron absorber from storage to transportation is necessary.

Thermal

Observation-1: On page A.3-82 of the application, the applicant stated that no fire test is performed and instead, the fire conditions are simulated using a finite element model of the MP197HB TC based on the requirements in 10 CFR Part 71. The thermal test described in 10 CFR 71.73(c)(4) reads as follows: "Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800°C (1475°F) for a period of 30 minutes, <u>or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800°C."</u>

The staff believes that the fire analysis provided in the application does not meet the conditions of the regulations (see underlined text above) because the applicant multiplied the fire temperature (800°C) by 0.9, which will reduce the radioactive heat transfer to the package during the fire. The applicant described the radiation heat transfer coefficient on Page A.3-82 of the application.

The fire analysis may be acceptable if the applicant performs the analysis at 1475°F and no multiplier is used (assuming fire emissivity equals 1.0). The applicant will need to resubmit modified SAR pages and updated fire thermal analysis files.

This information is required by the staff to verify compliance with 10 CFR 71.73(c)(4)

TN Response:

The analysis provided in the SAR used a fire temperature of 1475 °F with a fire emissivity of 0.9 to simulate total heat input equivalent to the thermal test described in 10 CFR 71.73 to the package. However, to evaluate the impact of using an emissivity of 1.0, a sensitivity analysis was performed and the results of this sensitivity analysis are documented in Appendix A.3.6.8. The effect of increasing the fire emissivity from 0.9 to 1.0 results in increasing the outermost cask component temperatures directly exposed to the fire but occurs only for a short period of time. The maximum TC component and the package content's temperatures remain well below the allowable limits.

Criticality

RSI-1: The application includes the use of incompatible cross section libraries between the MP-197 criticality analysis and the associated benchmarking calculations in Chapter 6.0 of the SAR. The applicant should revise the benchmarking analysis to use the same cross section libraries consistently for all the calculations performed.

The application includes an analysis of selected Commercial Reactor Critical (CRC) state points as part of the benchmarking analysis in Section A.6.3.2 of the SAR. This analysis includes k_{eff} calculations of various CRC state points using SCALE 4.x with KENO V.a and the 238-group ENDF/B-V cross section library. The criticality analysis of the MP-197 system uses the same code system, but with the 44-group ENDF/B-V cross section library. Section A.6.3.2.2 of the SAR states that the 44-group library consistently over-predicts k_{eff} in comparison to the 238-group library, and incorrectly refers to NUREG/CR-6686, "Experience with the SCALE Criticality Safety Cross-Section Libraries," for justification. This NUREG/CR provided a comparison of the capabilities of various cross section libraries used with the SCALE code system modules, for calculating the k_{eff} of various fissile systems, including uranium oxide and mixed uranium and plutonium oxide systems. However, NUREG/CR-6686 did not compare the performance of these cross-section libraries when calculating k_{eff} for burned fuel compositions. Additionally, ANSI/ANS-8.24-2007, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations," states that "The calculational methods and analysis techniques (e.g., albedos, variance reduction, *cross-section* processing) used to analyze the set of benchmarks shall be the same as those used to analyze the system or process to which the validation is applied." The CRC state point analysis should be revised to use the 44-group cross section library, or the package criticality analysis should be revised to use the cross section library used consistently in the benchmarking analysis.

This information is needed in order for the staff to proceed with a technical review to confirm that the package design meets the criticality safety requirements of 10 CFR 71.55 and 71.59.

TN Response:

TN has performed an evaluation to demonstrate that the results for the 44-group cross section library are conservative. This is achieved by a direct comparison of the CRC cases with both the 44-Group and 238-Group cross section libraries using SCALE 5.0.

Chapter A.6, Section A.6.3.2.2 is revised to include these analyses. Table A.6-13 is modified to include these results. Incorporating the CRC results calculated using TN computers results in a change in the USL from 0.9373 to 0.9370. Table in Section A.6.1.3, Table A.6-14 and Table A.6-15 are modified to include these results. Changes are also extended to Appendix A.6.5.4 through Appendix A.6.5.7 of Chapter A.6 to reflect the modified USL.

RSI-2: The applicant did not provide a technical basis for crediting greater than 50 GWd/MTU burnup of the spent fuel contents to be transported in the Model No. MP-197.

Division of Spent Fuel Storage and Transportation Interim Staff Guidance 8 (ISG-8), Rev. 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," states that the licensing basis for burnup credit should be limited to "UO_2 fuel irradiated in a PWR to an assembly-average burnup value up to 50 GWd/MTU." The application should be revised to only credit fuel burnup up to 50 GWd/MTU, or to provide a specific technical

analysis, with supporting data, to justify crediting greater than this amount of burnup.

This information is needed in order for the staff to proceed with a technical review to confirm that the package design meets the criticality safety requirements of 10 CFR 71.55 and 71.59.

TN Response:

The SAR is revised to include burnup credit only for fuel assemblies not exceeding 50 GWD/MTU. Changes are made to Chapter A.6, Section A.6.13.

The depletion and criticality analysis in the SAR is modified to accommodate this change. Chapter A.6, Section A.6.4.B and Table A.6-3 are modified to include the changes in the depletion calculations. The criticality calculations in Appendix A.6.5.4, Appendix A.6.5.5 and Appendix A.6.5.7 are modified to include these changes.

The burnup credit loading curves in Chapter A.1, Appendix A.1.4.3 through A.1.4.7 are modified to include these changes.

RSI-3: The applicant did not include a description or justification of the methodology used to apply burnup credit to damaged fuel configurations in the 24PTH, 32PTH/32PTH1, and 37PTH canisters.

ISG-8, Rev. 2 states: "The recommendations that follow are applicable to intact fuel. If burnup credit is requested for damaged fuel (basically intact, not debris), the recommendations of this guidance should be applied, as appropriate, to account for uncertainties that can be associated with the damaged fuel and establish an isotopic inventory and assumed fuel configuration for normal and accident conditions that bounds the uncertainties." The applicant should revise the criticality analysis to address the burnup credit assumptions for damaged furl configurations.

This information is needed in order for the staff to proceed with a technical review to confirm that the package design meets the criticality safety requirements of 10 CFR 71.55 and 71.59.

TN Response:

Damaged fuel assemblies are required to be handled by normal means at the time of loading. They do not contain any debris material or contain loose rods. Therefore, the isotopic inventory is identical for intact and damaged fuel assemblies. The criticality analysis for damaged fuel considers the most reactive credible configurations during normal and accident conditions (rod pitch variation, rod shear etc). All the geometrical uncertainties are accounted for by assuming bounding configurations for damaged fuel assemblies. Therefore, no other uncertainties need to be addressed for the geometry or isotopic inventory for damaged assemblies.

The criticality analysis is revised (Chapter A.6, Section A.6.2.5.2) to provide further clarification on the assumptions employed to describe the material composition of the damaged fuel assemblies.

Observation-1: The application does not discuss the applicability of a one-dimensional depletion code (SAS2H) for modeling the irradiation conditions of the fuel to be transported in the Model No. MP-197, as well as the radiochemical assay samples.

TN Response:

The benchmark calculations with isotopic assay data documented in Section A.6.3.2.1 provide the necessary justifications and associated limitations of the SAS2H code.

Observation-2: The use of BONAMI/NITAWL for cross-section processing in the criticality analysis can be impacted by improper treatment of cases with significant resonance overlap, leading to significant errors in k_{eff} determination. The application does not discuss the applicability of this code for modeling spent fuel compositions in the criticality analysis. The applicant should revise the application to include this discussion, or consider using CENTRM/PMC (available in later versions of the SCALE code system), which provides a more robust cross-section treatment with little computational penalty.

TN Response:

The benchmark calculations with critical experiments documented in Section A.6.3.2.2 provide the necessary justifications and associated limitations of the BONAMI/NITAWL treatment.

Observation-3: The application does not discuss the solver method of the DARWIN isotopic depletion code. This information will be useful to the NRC staff in determining the applicability of the code-to-code comparison in support of the MP-197 benchmarking analysis.

TN Response:

DARWIN methodology is briefly discussed in Chapter A.6, Section A.6.3.2.1. Reference 26 Chapter A.6, Section A.6.4 of the SAR describes the DARWIN methodology in detail. Further, it is also discussed in enclosure 10c of the Amendment application. A new reference (reference 28) is included in Chapter A.6, Section A.6.4 of the SAR. Enclosure 7 to this submittal provides a copy of this reference to support staff review.

Observation-4: The HTC benchmark isotopes are well known, and fabricated to correspond to a single, homogenous burnup of 37.5 GWd/MTU. The applicant should demonstrate the applicability of this indirect benchmark to the entire burnup range (0 - 60 GWd/MTU) for the package.

TN Response:

Chapter A.6, Section A.6.3.2.2 is modified to include a discussion on the applicability of the criticality benchmarks utilized to cover the entire range of burnups credited for the MP197HB package.

Shielding

RSI-1: Provide complete fuel assembly and non-fuel hardware contents design data, including geometric specifications, material specifications, and actual maximum fuel weight for all contents specified in the application. Also, demonstrate that the B&W 15x15 Mark B10 and GE-2, 3 7x7 Type G2A fuel assemblies provide bounding neutron and gamma sources for radiation shielding analysis for all fuel and hardware contents to be shipped using the TN MP197HB packaging system.

The staff was unable to find complete specifications of the design parameters for the proposed spent fuel contents for each of the DCSs proposed to be shipped in the MP197HB. Missing information includes: 1) fuel rod geometry data, such as rod pitch, rod diameter, clad thickness, fuel pellet diameter, 2) assembly configuration, 3) cladding material, and 4) assembly hardware (e.g. spacers and end fittings) material specifications, such as material type, mass and cobalt impurity levels. Without this data, the staff will not be able to confirm the spent fuel source terms.

Further, on page A.5-1, the application states: "For DSCs loaded with irradiated PWR and BWR fuel, the B&W 15x15 Mark B10 and the GE-2,3 7x7 Type G2A fuel assembly contains the maximum heavy metal weight in their type, nearly 490 and 198 kgU, respectively. Because of this, they result in bounding neutron and gamma source terms for PWR and BWR type of assemblies, respectively. Therefore B&W 15x15 Mark B-10 and the GE-2, 3 7x7 are identified as the Design Basis (DB) PWR and BWR Fuel Assembly (FA) in the shielding evaluation of the MP197HB transportation package, respectively." Staff notes that all assemblies in a given DSC are listed with the same uranium mass loading, resulting in multiple assembly types having identical maximum uranium mass loadings for both PWR and BWR fuel. Therefore, it is not clear that the selected assembly types are bounding for BWR and PWR fuel. If all assembly types may have the maximum uranium mass loading, then the other assembly parameters, not currently provided in the application, will determine which assembly types are bounding for PWR and BWR fuel.

In addition, the application indicates that control rod blades, BWR local power range monitors, BWR fuel channels, BWR poison curtains, PWR burnable poison rod assemblies, PWR and BWR reactor vessel and internals, thimble plug assemblies, control rod assemblies, control rod cluster assemblies, axial power shaping rods, orifice rod assemblies, vibration suppression inserts, irradiated/unirradiated stainless steel replacement rods in reconstituted fuel assemblies, neutron source assemblies, and neutron sources, are proposed contents for the cask. Partial length shielding assemblies also are proposed contents for the package. The staff was unable to find any specific information about any of the components listed above. The application should describe the materials composing these components, including the masses, assumed cobalt impurity levels, and basis for the impurity levels, as well as the assumed irradiation history.

This information is needed in order for the staff to proceed with a technical review of the package shielding design to confirm compliance with 10 CFR 71.47.

TN Response:

The fuel assembly rod design parameters are already included in Chapter A.2, Table A.2.13.11-2 for the BWR fuel and Table A.2.13.11-3 for the PWR fuel, as part of the fuel assembly impact analyses. Further, the same data is also included in Chapter A.6, Table

A.6.5.1-3, Table A.6.5.1-53 and Table A.6.5.2-2 for BWR fuel and Table A.6.5.3-2 and Table A.6.5.4-4 for PWR fuel, as part of the criticality analyses.

The SAR is modified to provide additional details as follows:

- *all references to the other SAR tables containing fuel data is provided (Chapter A.5, Section A.5.1.1)*
- *a discussion regarding bounding MTU loadings and hardware information is included (Chapter A.5, Section A.5.1.1)*
- *the maximum allowable MTU for fuel designs other than B&W 15x15 and GE 7x7 is reduced (Chapter A.1, Appendix A.4.1 through Appendix A.1.4.9 and Chapter A.5, Section A.5.1.1)*

Shielding evaluation for two different types of payloads (spent fuel and irradiated/contaminated radioactive waste) is performed for the MP197HB cask as described in Chapter A.1, Section A.1.2 of the SAR. The spent fuel payload consists of various DSCs with BWR or PWR fuel assemblies and associated control components (CCs). The irradiated/contaminated radioactive waste payload consists of the waste canister with the irradiated non-fuel hardware (including reactor materials).

The SAR is modified to provide additional details as follows:

- *a detailed description of the various payloads employed in the shielding evaluation that includes the CCs, reconstituted fuel assemblies and partial length shield assemblies is provided (Chapter A.5, Section A.5.1.1)*
- *a description of the source term employed for the irradiated/contaminated radioactive source is provided (Chapter A.5, Section A.5.1.2).*
- *additional clarification for CCs (Chapter A.5, Section A.5.5.2)*
- *additional clarification for reconstituted fuel assemblies (Chapter A.5, Section A.5.5.3)*

Observation-1: In Section A.5.3.5, condition #7 (p. A.5-19), the application indicates a "protective screen" or "personnel barrier" is "expected" to be around the package during transportation. However, this doesn't appear to be mentioned in the operational procedures, unless the "transportation skid closure assembly" (section A.7.1.5.1, p. A.7-18) is the same component. The documentation should be clear as to whether the "protective screen" is required, and if so, its use should be covered in the procedures.

TN Response:

The "personnel barrier" is included in the MP197HB cask drawings shown in Chapter A.1, Appendix A.1.4.10, Section A.1.4.10.1. The loading procedures described in Chapter A.7, Section A.7.1.5.1, are modified to include the installation of the personnel barrier.

Observation-2: The meaning of note 3 of Table A.5-1 (p. A.5-106) is not clear. The note should clarify the location of the dose rates to which the note applies. The table should clarify whether the listed dose rates are the bounding dose rates for the cask.

TN Response:

Note 2 is applicable to all the dose rates (Top, Side and Bottom) for the 2m distance while Note 3 is specific to all the payloads except the irradiated/contaminated waste canister and is calculated at 2m from the vehicle edge.

Observation-3: On page A.5-4, the application states: "The design basis radiological sources for NCT and HAC are due to DB FA irradiated at a constant specific power of 12.4 and 15.8 MW/assembly to a bundle average burnup of 55,000 and 70,000 MWd/MTU respectively." The applicant further discussed the reason for being able to use SAS2H beyond its validated range and concluded that the expected error is about 11%. It is not clear if this uncertainty has been included in the dose rate calculations. The application should describe how this uncertainty has been accounted for in the dose rate calculations.

TN Response:

The SAR Chapter A.5, Section A.5.2 is modified to include the following discussion.

The uncertainty value of 10% is an uncertainty in the ability of the SAS2H code to predict the isotopic concentration of nuclides in the fuel. In many cases, this results in SAS2H over-predicting the quantity of certain fission product or actinide isotope, thereby resulting in a conservative prediction of source terms. These benchmarks demonstrate that the neutron spectrum calculated by SAS2H during depletion is appropriate for the purpose of source term calculations. An uncertainty has not been applied in the dose rate calculations.

Observation-4: On pages A.5-107 and A.5-109, the application presents estimated dose rates at various locations of the MP-197 HB cask under normal conditions of transport and hypothetical accident conditions. The notes to Tables A.5-3 and A.5-5 indicate that these dose rates are averaged over very large areas. Given the dose rates for the evaluated rail car lengths and the size of the area over which they are averaged, it appears that personnel will need to wear radiation dosimetry devices in order to ensure compliance with 10 CFR 71.47(b)(4). The application should discuss this need for dosimetry or show with a more reasonably sized area (over which dose rates are averaged) the rail car lengths for which dosimetry is not needed at occupied locations.

TN Response:

The dose rates at occupied locations (averaged over an area that is slightly larger than the package diameter) during NCT for the MP197HB with the spent fuel (DSCs) payload is shown in Table A.5-3. The NCT maximum dose rates as a function of distance are shown in Table A.5-21 and Table A.5-22 for Bottom and Top ends respectively. As can be seen from these results, the average dose rates are a reasonable representation of total exposure during transport – most importantly, the peak to average dose rate factor is between 1.2 and 1.4 at dose rates in the vicinity of 2 mrem/hour. These results demonstrate that the maximum dose rates will be below 2 mrem/hour at distances greater than 3.3 m from the outer surface of the impact limiter at the cask bottom. This information is added to Section A.5.4.1 and Note 1 of Table A.5-3.

The dose rates at occupied locations (averaged over an area that is slightly larger than the package diameter) during NCT for the MP197HB with the irradiated/contaminated

radioactive waste payload is shown in Table A.5-5. The NCT maximum dose rates as a function of distance are shown in Table A.5-27 at both the Bottom and Top end locations. As can be seen from these results, the average dose rates are a reasonable representation of total exposure during transport – most importantly, the peak to average dose rate factor is between 1.2 and 1.3 at dose rates in the vicinity of 2 mrem/hour. These results demonstrate that the maximum dose rates will be below 2 mrem/hour at distances greater than 5.5 m from the outer surface of the impact limiter at the cask bottom. This information is added to Section A.5.4.2 and Note 1 of Table A.5-5.

Observation-5: The applicant should clarify the location of the maximum total dose rate at the package's side surface given in Table A.5-1. The dose rate value in the same table for the irradiated waste/contaminated waste container appears to be taken at the surface of the personnel barrier based upon Table A.5-28. If the location is the same for the maximum total dose rate from a DSC, the appropriate limit would be 200 mrem/hr, since (assuming that the MP-197 HB is transported on an open vehicle) the limit for the top of the enclosure is 200 mrem/hr (which is along the side of the MP-197 HB). Further, it is not clear from the other tables (Tables A.5-21through A.5-26) where the listed maximum value (191.2 mrem/hr) occurs. Depending upon the correct limit, this surface side dose rate may be the most restrictive limit for which response functions should be determined.

TN Response:

This information is added to SAR Chapter A.5, Section A.5.5.1 (at the end of the section).

List of Changed NUHOMS®-MP197 Safety Analysis Report Pages, Revision 6 (for the Proprietary Version)

SAR Page	Replace Page, Insert New Page, or Remove Page?	Applicable RSI or Observation	Reason for Change
Cover Sheet	Replace Page	none	SAR overall revision changed to Revision 6.
Appendix A TOC Page 7	Replace Page	none	Table of Contents updated to match changes.
Appendix A TOC Page 8	Replace Page	none	Table of Contents updated to match changes.
Page A.1.4.2-8	Replace Page	none	SAR Table A.1.4.2-6, 5th bullet - changed "less than 2.0 or greater 5.0" to "less than 1.1 or greater than 5.0" to match the table.
Page A.1.4.3-5	Replace Page	Structural RSI No. 1 Criticality RSI No. 3	Per the RSI responses.
		none	Changed "6.5 years" to "10 years" to be consistent with the criticality analysis.
Page A.1.4.3-6	Replace Page	Structural RSI No. 1 Criticality RSI No. 3	Per the RSI responses.
		none	The spelling of "accommodate" is corrected.
Page A.1.4.3-8	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.1.4.3-11	Replace Page	none	On Table A.1.4.3-6, the spelling of "Matrix" is corrected.
Page A.1.4.3-13	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.4-5	Replace Page	Structural RSI No. 1 Criticality RSI No. 3	Per the RSI response.
Page A.1.4.4-6	Replace Page	Structural RSI No. 1 Criticality RSI No. 3	Per the RSI response.
Page A.1.4.4-10	Replace Page	none	Table notes for Table A.1.4.4-5, bullet 5, changed "62" to "60" to match the table.
Page A.1.4.4-13	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.4-14	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.5-5	Replace Page	Structural RSI No. 1 Criticality RSI No. 3	Per the RSI response.
Page A.1.4.5-8	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.1.4.5-13	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.5-14	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.5-15	Replace Page	Criticality RSI No. 2	Per the RSI response.

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SAR Page	Replace Page, Insert New Page, or Remove Page?	Applicable RSI or Observation	Reason for Change
Page A.1.4.6-5	Replace Page	Structural RSI No. 1 Criticality RSI No. 3	Per the RSI response.
Page A.1.4.6-8	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.1.4.6-11	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.6-12	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.1.4.7-3	Replace Page	Structural RSI No. 1 Shielding RSI No. 1 Criticality RSI No. 3	Per the RSI responses.
		none	Editorial correction to Table A.1.4.7-1 Note (1): changed "Table 1.4.7-4" to "Table A.1.4.7-4".
Page A.1.4.7-4	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.1.4.8-1	Replace Page	none	Correction in Section A.1.4.8.1, first paragraph: deleted "As shown in Table A.1.4.8-1, " Also changed "8" to "4" in the third bullet to be consistent with the remainder of the SAR.
Page A.1.4.8-2	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.1.4.8-5	Replace Page	Structural RSI No. 1 Shielding RSI No. 1 Criticality RSI No. 3	Per the RSI responses.
Page A.1.4.8-6	Replace Page	Structural RSI No. 1 Shielding RSI No. 1	Per the RSI response.
		none	Editorial correction to Table A.1.4.8-2 Note (1): changed "Table 1.4.8-6/1.4.8-7" to "Table A.1.4.8-6/A.1.4.8-7".
Page A.1.4.8-7	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.1.4.9-3	Replace Page	Structural RSI No. 1 Shielding RSI No. 1 Criticality RSI No. 3	Per the RSI responses.
Page A.1.4.9-4	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.2.13.11-i	Replace Page	Structural RSI-1	Table of Contents updated.
Page A.2.13.11-ii	Replace Page	Structural RSI-1	Table of Contents updated.
Page A.2.13.11-12 through 22	Replace Pages	Structural RSI-1	Brittle fracture evaluations are removed.

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SAR Page	Replace Page, Insert New Page, or Remove Page?	Applicable RSI or Observation	Reason for Change
Page A.2.13.11-54	Replace Page	Structural RSI-1	Brittle fracture evaluations are removed.
Page A.2.13.11-55 through 64	Remove Pages	Structural RSI-1	Brittle fracture evaluations are removed.
Page A.3-i	Replace Page	Thermal Observation	Updated table of contents for new Appendix A.3.6.8.
Page A.3-82	Replace Page	Thermal Observation	Added mention of new Appendix A.3.6.8.
Page A.3-139	Replace Page	Thermal Observation	New Appendix A.3.6.8 begins on this page.
Page A.3-139a	Insert New Page	Thermal Observation	New Appendix A.3.6.8 continues on this new page.
Page A.3-139b	Insert New Page	Thermal Observation	New Appendix A.3.6.8 continues on this new page.
Page A.5-i	Replace Page	none	Table of Contents updated due to changes in Chapter A.5.
Page A.5-2	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.5-3	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.5-3a	Insert New Page	Shielding Observation No. 5	Per the observation response.
Page A.5-3b	Insert New Page	Shielding RSI No. 1	Per the RSI response.
Page A.5-4	Replace Page	none	Information shifted; no changes show on the page.
Page A.5-5	Replace Page	none	Information shifted; no changes show on the page.
Page A.5-6	Replace Page	Shielding Observation No. 3	Per the observation response.
Page A.5-21	Replace Page	Shielding Observation No. 4	Per the observation response.
Page A.5-22	Replace Page	Shielding Observation No. 4	Per the observation response.
Page A.5-27	Replace Page	Shielding RSI No. 1	Per the RSI response.
Page A.5-107	Replace Page	Shielding Observation No. 4	Per the observation response.
Page A.5-109	Replace Page	Shielding Observation No. 4	Per the observation response.
Page A.6-2	Replace Page	Criticality RSI No. 1 Criticality RSI No. 2	Per the RSI responses.
Page A.6-7	Replace Page	Criticality RSI No. 2	Per the RSI response.

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SAR Page	Replace Page, Insert New Page, or Remove Page?	Applicable RSI or Observation	Reason for Change
Page A.6-12	Replace Page	Criticality RSI No. 3	Per the RSI response.
Page A.6-14	Replace Page	none	Corrected the spelling of "actinides".
		Criticality Observation No. 3	Per the observation response.
Page A.6-20	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6-20a	Insert New Page	Criticality RSI No. 1 Criticality Observation No. 4	Per the RSI response and per the observation response.
Page A.6-21	Replace Page	Criticality Observation No. 4	Per the observation response.
Page A.6-24	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6-29	Replace Page	Criticality RSI No. 1 Criticality RSI No. 2	Per the RSI response.
Page A.6-31	Replace Page	Criticality Observation No. 3	Per the observation response.
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Page A.6-66	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6-66a	Insert New Page	Criticality RSI No. 1	Per the RSI response.
Page A.6-67	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6-68	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.4-15	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.4-16	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.4-48	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-49	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-50	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-52	Replace Page	Criticality RSI No. 2	Per the RSI response.

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SAR Page	Replace Page, Insert New Page, or Remove Page?	Applicable RSI or Observation	Reason for Change
Page A.6.5.4-53	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-54	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-55	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-56	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-57	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-58	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-59	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-62	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.4-63	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.5-15	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.5-16	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.5-37	Replace Page	none	Corrected values "20.0" and "18.0" to "32.0" and "28.8", respectively.
Page A.6.5.5-43	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.5-46	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.5-47	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.6-13	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.7-14	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.7-15	Replace Page	Criticality RSI No. 1	Per the RSI response.
Page A.6.5.7-37	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.7-38	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.7-39	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.7-40	Replace Page	Criticality RSI No. 2	Per the RSI response.

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SAR Page	Replace Page, Insert New Page, or Remove Page?	Applicable RSI or Observation	Reason for Change
Page A.6.5.7-41	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.7-43	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.6.5.7-44	Replace Page	Criticality RSI No. 2	Per the RSI response.
Page A.7-18	Replace Page	Shielding Observation No. 1	Per the observation response.

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SAR Page	Replace Page, Insert New Page, or Remove Page?
Cover Sheet	Replace Page
Appendix A Table of Contents Pages 7 and 8	Replace Pages
Page A.1.4.2-8	Replace Page
Page A.1.4.3-5 and -6	Replace Pages
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Page A.1.4.5-8	Replace Page
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Page A.5-3b	Insert New Page
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Page A.5-107	Replace Page
Page A.5-109	Replace Page
Page A.6-2	Replace Page
Page A.6-7	Replace Page

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SAR Page	Replace Page, Insert New Page, or Remove Page?
Page A.6-12	Replace Page
Page A.6-14	Replace Page
Page A.6-20	Replace Page
Page A.6-20a	Insert New Page
Page A.6-21	Replace Page
Page A.6-24	Replace Page
Page A.6-29	Replace Page
Page A.6-31	Replace Page
Page A.6-42	Replace Page
Page A.6-65 and -66	Replace Pages
Page A.6-66a	Insert New Page
Page A.6-67 and -68	Replace Pages
Page A.6.5.4-15 and -16	Replace Pages
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Enclosure 6 to TN E-28237

Replacement and New NUHOMS®-MP197 Safety Analysis Report Pages, Revision 6

(for the Non-proprietary version)

NON-PROPRIETARY

AREVA

TRANSNUCLEAR INC.

NUHOMS®-MP197 TRANSPORT PACKAGING SAFETY ANALYSIS REPORT

Revision *6*

TRANSNUCLEAR INC.
7135 Minstrel Way • Suite 300 • Columbia, MD 21045

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Table A.1.4.2-6
PWR Fuel Qualification Table for NUHOMS®-32PT DSC
(Minimum required years of cooling time after reactor core discharge)

BU GWD/ MTU	Assembly Average Initial U-235 Enrichment, wt %																																					
	1.1	1.2	1.4	1.6	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
15	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
20	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
25	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
28	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
30	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
32	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
34	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
36	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
38	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
39	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
40	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
41	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
42	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
43	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
44	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
45	11.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 year for fuel assemblies in the 16 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than *1.1* or greater *than* 5.0 wt.% U-235 is unacceptable for Transport.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 10-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after a 10-year year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- Even though cooling times less than 15 years are shown in this table, the minimum cooling time requirement for criticality from Table A.1.4.2.7 for transportation is 15 years.

Table A.1.4.3-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-24PTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged or failed unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.3-4. Equivalent reload fuel manufactured by same or other vendors but enveloped by the design characteristics listed in Table A.1.4.3-4 is also acceptable.
Damaged Fuel	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel *assembly needs to be handled by normal means.* *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Failed Fuel	Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly can not be handled by normal means. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 1682 lbs.
Partial Length Shield Assemblies (PLSAs)	WE 15x15 class PLSAs with following characteristics are authorized: • Maximum burnup, 40 GWd/MTU • Minimum cooling time, *10* years • Maximum Decay Heat, 900 Watts
Reconstituted Fuel Assemblies: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO_2 Rods and/or Unirradiated Stainless Steel Rods and/or Zr Rods or Zr Pellets	 4 10 24

Table A.1.4.3-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-24PTH DSC
(concluded)

Control Components (CCs)	• Up to 24 CCs are authorized for storage in 24PTH-S, 24PTH-L, and 24PTH-S-LC DSCs. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. • Design basis thermal and radiological characteristics for the CCs are listed in Table A.1.4.3-3.
Nominal Assembly Width for Intact and Damaged Fuel Assemblies Only	8.536 inches
No. of Intact Assemblies	≤24
No. and Location of Damaged Assemblies	Up to 12 damaged fuel assemblies. Balance may be intact fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Damaged fuel assemblies are to be placed in Locations A and/or B as shown in Figure A.1.4.3-6. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
No. and Location of Failed Assemblies	Up to 8 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Failed fuel assemblies are to be placed in Location A as shown in Figure A.1.4.3-6. Failed fuel assembly/fuel debris is to be encapsulated in an individual Failed Fuel Can (FFC) provided with a welded bottom closure and a removable top closure.
Maximum Assembly plus CC Weight	1682 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Fuel Assembly Average Burnup and minimum Cooling Time(1)	Per Table A.1.4.3-5; Table A.1.4.3-8 and decay heat and burnup credit restrictions below.
Maximum Decay Heat(1) Limits for Zones 1, 2, 3, and 4 Fuel	Per Figure A.1.4.3-1 or Figure A.1.4.3-2 or Figure A.1.4.3-3 or Figure A.1.4.3-4 or Figure A.1.4.3-5.
Decay Heat(1) per DSC	Type 1 Basket ≤ 26.0 kW for 24PTH-S and 24PTH-L DSCs with decay heat limit for Zones 1, 2, 3 and 4 as specified in Figure A.1.4.3-1, or Figure A.1.4.3-2, Figure A.1.4.3-3 or Figure A.1.4.3-4.
	Type 2 Basket Same as Type 1 Basket except ≤26.0 kW/DSC and ≤ 1.3 kW/fuel assembly for 24PTH-S and 24PTH-L DSCs. ≤ 24.0 kW for 24PTH-S-LC DSC with decay heat limits as≤ 24.0 kW for 24PTH-S-L DSC (Type 2 Basket) specified in Figure A.1.4.3-5.
Burnup Credit Restrictions(1)	Per Table A.1.4.3-8.

Notes:

(1) Minimum cooling time is the longer of that given in Table A.1.4.3-5; that calculated via the decay heat equation given in Table A.1.4.3-9 based on the restrictions provided in Figures A.1.4.3-1, A.1.4.3-2, A.1.4.3-3 or A.1.4.3-4; and Table A.1.4.3-8.

Table A.1.4.3-4
PWR Fuel Assembly Design Characteristics for the NUHOMS®-24PTH DSC

Assembly Class		B&W 15x15	WE 17x17	CE 15x15	WE 15x15	CE 14x14	WE 14x14
Max Unirradiated Length (in)(1)	24PTH-S	165.75	165.75	165.75	165.75	165.75	165.75
	24PTH-L	171.93	171.93	171.93	171.93	171.93	171.93
	24PTH-S-LC	171.93	NA(3)	NA(3)	NA(3)	NA(3)	NA(3)
Fissile Material		UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/Assembly(2)		0.49	0.482	0.482	0.482 (4)	0.482	0.482
Maximum Number of Fuel Rods		208	264	216	204	176	179
Maximum Number of Guide/ Instrument Tubes		17	25	9	21	5	17

(1) Maximum Assembly + Control Component Length (unirradiated)
(2) The maximum MTU/assembly is based on the shielding analysis. The listed value is higher than the actual.
(3) Not Authorized.
(4) The maximum MTU/assembly for WE 15x15 PLSA = 0.33.

Table A.1.4.3-6
B10 Specification for the NUHOMS®-24PTH Poison Plates

NUHOMS®-24PTH DSC Basket Type [1]	Minimum B10 Areal Density, gm/cm^2	
	Natural or Enriched Boron Aluminum Alloy / Metal Matrix Composite (MMC)	Boral®
1A or 2A	.007	.009
1B or 2B	.015	.019
1C or 2C	.032	.040

Notes:
[1] Basket Type 1 contains aluminum inserts in the R45 transition rails; Type 2 does not contain aluminum inserts.

Table A.1.4.3-8
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-24PTH

	WE 17x17, WE 15x15 and BW 15x15 assembly classes		
Enrichment (wt. % U-235)	**Type A**	**Type B**	**Type C**
1.55	fresh	-	-
1.65	-	fresh	-
1.80	-	-	fresh
	Minimum Burnup (GWD/MTU), 30 Years decay		**Minimum Burnup (GWD/MTU), 15 Years decay**
2.00	16	12	8
2.25	19	17	16
2.50	22	19	18
2.75	27	22	20
3.00	30	26	24
3.25	34	30	29
3.50	36	33	32
3.75	41	36	35
4.00	*42*	39	37
4.20	*44*	*41*	40
4.40	*46*	*42*	43
4.60	*50*	*44*	*44*
4.80		*46*	*46*
5.00		*49*	*48*

Table A.1.4.4-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-32PTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged Westinghouse 17x17 (WE 17x17), Westinghouse 15x15 (WE 15x15), Combustion Engineering 16x16 (CE 16x16), and/or Combustion Engineering 14x14 (CE 14x14) class PWR fuel assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.4-3. Reload fuel manufactured by the same or other vendors but bounded by the design characteristics listed in Table A.1.4.4-3 is also acceptable.
Damaged Fuel	Damaged PWR fuel assemblies are assemblies with missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of the damage is to be limited such that a fuel assembly needs to be handled by normal means. *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Reconstituted Fuel Assemblies: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO2 Rods and/or Unirradiated Stainless Steel Rods and/or Zr Rods or Zr Pellets	4 10 32
Control Components (CCs)	• Up to 32 CCs are authorized for storage in 32PTH DSC. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. • Design basis thermal and radiological characteristics for the CCs are listed in Table A.1.4.4-4.
No. of Intact Assemblies	≤32

Table A.1.4.4-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-32PTH DSC
(concluded)

No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies with the balance intact fuel assemblies, or dummy assemblies. Damaged fuel assemblies are to be placed in the center 16 locations as shown in Figure A.1.4.4-2. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus CC Weight	1585 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Fuel Assembly Average Burnup and minimum Cooling Time(1)	Per Table A.1.4.4-5; Table A.1.4.4-8 and decay heat and burnup credit restrictions below.
Decay Heat(1)	Per Figure A.1.4.4-1
Burnup Credit Restrictions(1)	Per Table A.1.4.4-8.

Notes:
(1) Minimum cooling time is the longer of that given in Table A.1.4.4-5; that calculated via the decay heat equation given in Table A.1.4.4-7; based on the restrictions provided in Figures A.1.4.4-1; and Table A.1.4.4-8.

Notes: Table A.1.4.4-5:

- BU = Assembly average burnup.
- Use burnup and enrichment to look-up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 0.3 or greater than 5.0 wt.% U-235 is unacceptable for Transport.
- Fuel with a burnup greater than *60* GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 10-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after 10-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- Even though cooling times less than 15 years are shown in this table, the minimum cooling time requirement for criticality from Table A.1.4.4-8 for transportation is 15 years.

Table A.1.4.4-8
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-32PTH

	WE 17x17, and WE 15x15 assembly classes					
Enrichment (wt. % U-235)	**Type C**	**Type D**	**Type E**	**Type A**	**Type B**	**Type C**
1.50	-	-	-	fresh	-	-
1.60	-	-	-	-	fresh	-
1.65	fresh	-	-	-	-	fresh
1.75	-	fresh	-	-	-	-
1.80	-	-	fresh	-	-	-
	Minimum Burnup (GWD/MTU), 15 years decay			**Minimum Burnup (GWD/MTU), 30 years decay**		
2.00	14	12	8	20	14	12
2.25	18	17	15	20	18	17
2.50	22	19	20	26	21	20
2.75	26	22	20	30	25	24
3.00	30	27	24	33	29	28
3.25	34	31	28	36	33	31
3.50	37	34	32	41	35	34
3.75	41	37	34	44	40	37
4.00	*44*	40	37	*46*	*42*	40
4.20	*46*	45	40	*48*	*44*	43
4.40	*48*	*46*	43	*50*	*46*	*44*
4.60		*48*	*44*		*48*	*46*
4.80		*50*	*46*		*50*	*48*
5.00			*48*			*50*

Table A.1.4.4-8
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-32PTH
(concluded)

	CE 14x14 and CE 16x16 assembly classes			
Enrichment (wt. % U-235)	**Type C**	**Type D and Type E**	**Type A**	**Type B**
1.65	-	-	fresh	-
1.75	-	-	-	fresh
1.80	fresh	-	-	-
1.90	-	fresh	-	-
	Minimum Burnup (GWD/MTU), 15 years decay		**Minimum Burnup (GWD/MTU), 30 years decay**	
2.00	12	8	18	12
2.25	17	15	19	17
2.50	19	18	23	19
2.75	22	19	27	22
3.00	26	24	31	25
3.25	30	26	33	29
3.50	32	30	35	31
3.75	34	32	38	33
4.00	37	34	41	35
4.20	40	36	*42*	38
4.40	42	39	*44*	41
4.60	45	41	*46*	43
4.80	47	43	*48*	*44*
5.00	49	45	*50*	*45*

Notes:

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)
- Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.
- Extrapolation shall not be performed to determine burnup requirements.
- The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.

Table A.1.4.5-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-32PTH1 DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, WE 14x14 and CE 16x16 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.5-4. Reload fuel manufactured by the same or other vendors but enveloped by the design characteristics listed in Table A.1.4.5-4 is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for transport.
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is able to be handled by normal means. *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO_2 Rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	4 10 32
Control Components (CCs)	• Up to 32 CCs are authorized for storage in 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), and Neutron Source Assemblies (NSAs), and Neutron Sources • Design basis thermal and radiological characteristics for the CCs are listed in Table A.1.4.5-3.
No. of Intact Assemblies	≤32
No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies. Balance may be intact fuel assemblies, or dummy assemblies which are authorized for storage in 32PTH1 DSC. Damaged fuel assemblies are to be placed in the center 16 locations as shown in Figure A.1.4.5-1, Figure A.1.4.5-2 and Figure A.1.4.5-3. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus CC Weight	1715 lbs

Table A.1.4.5-4
PWR Fuel Assembly Design Characteristics for the NUHOMS®-32PTH1 DSC

Assembly Class		**B&W 15x15**	**WE 17x17**	**CE 15x15**	**WE 15x15**	**CE 14x14**	**WE 14x14**	**CE 16x16**
Max Unirradiated Length (in)(1)	32PTH1-S	162.6	162.6	162.6	162.6	162.6	162.6	162.6
	32PTH1-M	170.0	170.0	170.0	170.0	170.0	170.0	170.0
	32PTH1-L	178.3	178.3	178.3	178.3	178.3	178.3	178.3
Fissile Material		UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/Assembly(2)		0.49	0.482	0.482	0.482	0.482	0.482	0.482
Maximum Number of Fuel Rods		208	264	216	204	176	179	236
Maximum Number of Guide/ Instrument Tubes		17	25	9	21	5	17	5

Notes:
(1) Maximum Assembly + Control Component Length (unirradiated)
(2) The maximum MTU/assembly is based on the shielding analysis. The listed value is higher than the actual.

Table A.1.4.5-8
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1

	WE 17x17, WE 15x15 and BW 15x15 assembly classes					
Enrichment (wt. % U-235)	**Type C**	**Type D**	**Type E**	**Type A**	**Type B**	**Type C**
1.50	-	-	-	fresh	-	-
1.60	-	-	-	-	fresh	-
1.65	fresh	-	-	-	-	fresh
1.75	-	fresh	-	-	-	-
1.80	-	-	fresh	-	-	-
	Minimum Burnup (GWD/MTU), 15 years decay			**Minimum Burnup (GWD/MTU), 30 years decay**		
2.00	14	12	8	20	14	12
2.25	18	17	15	20	18	17
2.50	22	19	20	26	21	20
2.75	26	22	20	30	25	24
3.00	30	27	24	33	29	28
3.25	34	31	28	36	33	31
3.50	37	34	32	41	35	34
3.75	41	37	34	44	40	37
4.00	*44*	40	37	*46*	*42*	40
4.20	*46*	45	40	*48*	*44*	43
4.40	*48*	*46*	43	*50*	*46*	*44*
4.60		*48*	*44*		*48*	*46*
4.80		*50*	*46*		*50*	*48*
5.00			*48*			*50*

Table A.1.4.5-8
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1
(continued)

	CE 14x14, CE 15x15 and CE 16x16 assembly classes			
Enrichment (wt. % U-235)	**Type C**	**Type D Type E**	**Type A**	**Type B**
1.65	-	-	fresh	-
1.75	-	-	-	fresh
1.80	fresh	-	-	-
1.90	-	fresh	-	-
	Minimum Burnup (GWD/MTU), 15 years decay		**Minimum Burnup (GWD/MTU), 30 years decay**	
2.00	12	8	18	12
2.25	17	15	19	17
2.50	19	18	23	19
2.75	22	19	27	22
3.00	26	24	31	25
3.25	30	26	33	29
3.50	32	30	35	31
3.75	34	32	38	33
4.00	37	34	41	35
4.20	40	36	*42*	38
4.40	42	39	*44*	41
4.60	45	41	*46*	43
4.80	47	43	*48*	*44*
5.00	49	45	*50*	*45*

Table A.1.4.5-8
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1
(concluded)

	WE 14x14 assembly class		
Enrichment (wt. % U-235)	**Type B**	**Type C, D, or E**	**Type A**
1.70	-	-	fresh
1.85	fresh	-	-
1.90	-	fresh	-
	Minimum Burnup (GWD/MTU) 15 years decay		**Minimum Burnup (GWD/MTU) 30 years decay**
2.00	7	7	10
2.25	15	14	16
2.50	17	16	18
2.75	19	18	20
3.00	23	21	25
3.25	26	24	28
3.50	30	29	32
3.75	33	32	34
4.00	35	34	37
4.20	38	36	40
4.40	40	39	42
4.60	43	41	*44*
4.80	*45*	44	*46*
5.00	*47*	46	*48*

Notes:

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted).
- Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt.% U-235 and 5.00 wt.% U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.
- Extrapolation shall not be performed to determine burnup requirements.
- The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.

Table A.1.4.6-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-37PTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged unconsolidated WE 17x17, CE 15x15, WE 15x15, CE 14x14, WE 14x14 and CE 16x16 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.6-4. Reload fuel manufactured by same or other vendors but enveloped by the design characteristics listed in Table A.1.4.6-4 is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is able to be handled by normal means. *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO_2 Rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	4 10 37
Control Components (CCs)	• Up to 37 CCs are authorized for storage in 37PTH-S, and 37PTH-M DSCs. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), and Neutron Source Assemblies (NSAs), and Neutron Sources • Design basis thermal and radiological characteristics for the CCs are listed in Table A.1.4.6-3.
No. of Intact Assemblies	≤37
No. and Location of Damaged Assemblies	Up to 4 damaged fuel assemblies. Balance may be intact fuel assemblies, or dummy assemblies which are authorized for storage in 37PTH DSC. Damaged fuel assemblies are to be placed in the four corner locations as shown in Figure A.1.4.6-1. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus CC Weight	1665 lbs for 37PTH-S 1625 lbs for 37PTH-M

Table A.1.4.6-4
PWR Fuel Assembly Design Characteristics for the NUHOMS®-37PTH DSC

Assembly Class		**WE 17x17**	**CE 15x15**	**WE 15x15**	**CE 14x14**	**WE 14x14**	**CE 16x16**
Max Unirradiated Length (in)(1)	37PTH-S	162.6	162.6	162.6	162.6	162.6	162.6
	37PTH-M	170.0	170.0	170.0	170.0	170.0	170.0
Fissile Material		UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/Assembly(2)		0.482	0.482	0.482	0.482	0.482	0.482
Maximum Number of Fuel Rods		264	216	204	176	179	236
Maximum Number of Guide/ Instrument Tubes		25	9	21	5	17	5

Notes:
(1) Maximum Assembly + Control Component Length (unirradiated)
(2) The maximum MTU/assembly is based on the shielding analysis. The listed value is higher than the actual.

Table A.1.4.6-6
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-37PTH

Enrichment (wt. % U-235)	**WE 17x17 and WE 15x15 assembly classes**	
1.50	-	
1.60	-	
1.65	fresh	
1.75	-	
1.80	-	
	Minimum Burnup (GWD/MTU), 15 years decay	**Minimum Burnup (GWD/MTU), 30 years decay**
2.00	14	14
2.25	18	18
2.50	22	20
2.75	26	23
3.00	30	28
3.25	34	32
3.50	37	34
3.75	41	38
4.00	*42*	41
4.20	*45*	*42*
4.40	*49*	*43*
4.60		*45*
4.80		*48*
5.00		

Table A.1.4.6-6
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-37PTH
(continued)

Enrichment (wt. % U-235)	**CE 14x14, CE 15x15 and CE 16x16 assembly classes**
1.65	-
1.75	-
1.80	fresh
1.90	-
	Minimum Burnup (GWD/MTU), 15 years decay
2.00	12
2.25	17
2.50	19
2.75	22
3.00	27
3.25	30
3.50	33
3.75	35
4.00	37
4.20	40
4.40	42
4.60	*44*
4.80	*45*
5.00	*47*

Table A.1.4.7-1
BWR Fuel Specification for Fuel to be Transported in the NUHOMS®-61BT DSC

PHYSICAL PARAMETERS: Fuel Design	Intact or damaged unconsolidated 7x7, 8x8, 9x9, or 10x10 intact BWR fuel assemblies manufactured by General Electric or Exxon/ANF or reload fuel manufactured by the same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.7-2.
Fuel Damage(3)	Damaged BWR fuel assemblies are 7x7 and 8x8 fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. *The extent of cladding damage in the fuel rods* is to be limited such that a fuel *assembly needs to be handled by normal means.* Damaged fuel may only be transported in the "Type C" NUHOMS®-61BT Canister. Damaged fuel is restricted to the 7x7 and 8x8 designs only. *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Channels	Fuel may be transported with or without fuel channels
No. of Intact Assemblies	≤61
No. and Location of Damaged Assemblies	Up to sixteen (16) damaged fuel assemblies may be accommodated in the four corner 2x2 compartment assemblies with endcaps installed on each end of the compartment.
Maximum Assembly plus fuel channel weight	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS(1): Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1.4.7-3
Fuel Assembly Average Burnup and minimum Cooling Time(1)	Per Table A.1.4.7-4 and decay heat restrictions below
Decay Heat(1)(2)	0.300 kW/Assembly calculated per Table A.1.4.7-5

(1) Minimum cooling time is the longer of that given in Table *A*.1.4.7-4; that calculated via the decay heat equation given in Table A.1.4.7-5 to meet the 0.300 kW/assembly limit.
(2) For FANP9 9x9-2 fuel assemblies, the maximum decay heat is limited to 0.21 kW/assembly.
(3) For damaged fuel assemblies, the maximum initial lattice average enrichment and maximum pellet enrichment is limited to 4.0 wt.% U-235 and 4.4 wt.% U-235 respectively.

Table A.1.4.7-2
BWR Fuel Assembly Design Characteristics (1) (2)

Transnuclear, ID	**7 x 7-49/0[5]**	**8 x 8-63/1[5]**	**8 x 8-62/2[5]**	**8 x 8-60/4[5]**	**8 x 8-60/1[5]**	**9 x 9-74/2**	**10x10-92/2**	**7x7 – 49/0[5]**	**7x7 48/1Z[5]**	**8x8 – 60/4Z[5]**	**9x9-79/2**
Fuel Type	GE1 GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12	ENC III-A	ENC III[3]	ENC Va & ENC Vb	FANP9 9x9-2
Nominal Width (in.) (excluding channels)	5.44	5.44	5.44	5.44	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Number of Fuel Rods	49	63	62	60	60	66 – Full 8 – Partial	78 – Full 14 – Partial	49	48	60	79
Number of Water Holes	0	1	2	4	1	2	2	0	1[4]	4[4]	2
Maximum Initial Uranium Content (kg)	*198*	*192*	*192*	*192*	*192*	*192*	*192*	*198*	*198*	*192*	*192*

(1) Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.
(2) Maximum fuel assembly unirradiated length is 176.2 inch.
(3) Includes ENC III-E and ENC III-F.
(4) Solid Zirc rods instead of water holes.
(5) May be transported as damaged fuel.

Appendix A.1.4.8
NUHOMS®-61BTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.8.4.

A.1.4.8.1 NUHOMS®-61BTH DSC Description

Each NUHOMS®-61BTH DSC consists of a DSC shell assembly and basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. The 61BTH DSC system consists of three design configurations, depending upon the type of fuel and heat load, as follows:

- 61BTH Type 1
- 61BTH Type 2
- 61BTHF, accommodates up to 4 Failed Fuel Cans with Failed Fuel

Table A.1.4.8-1 provides the overall lengths and outer diameters for each 61BTH DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in the axial direction). The 61BTH DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 61BTH DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 61BTH DSC drying operations are complete.

The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal conditions, the canister rests on four canister rails attached to the inside surface of the aluminum inner sleeve of the transport cask.

A.1.4.8.2 NUHOMS®-61BTH DSC Fuel Basket

The basket structure is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and outer diameters of the baskets, including the hold down rings, are provided in Table A.1.4.8-1. The details of the 61BTH fuel baskets are shown in the drawings in Section A.1.4.10.9 of Appendix A.1.4.10. The 61BTH baskets are designed to accommodate 61 intact, or up to 16 damaged with up to four (4) Failed Fuel Cans (FFCs) loaded with failed fuel with the remainder intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel structural boxes. The basket is laterally supported by the basket rails and the canister shell. The stainless steel basket rails are oriented parallel to the axis

of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

The failed fuel assemblies are to be placed in individual Failed Fuel Cans (FFCs). Each FFC is constructed of sheet metal and is provided with a welded bottom closure and a removable top closure which allows lifting of the FFC with the enclosed damaged assembly/debris. The FFC is provided with screens at the bottom and top to contain fuel debris and allow fill/drainage of water from the FFC during loading operations. The FFC is protected by the fuel compartment tubes and its only function is to confine the failed fuel.

A shear key, welded to the inner wall of the DSC, mates with a notch in one of the basket support rails to prevent the basket from rotating during normal operations. Also a hold down ring is installed above the basket to prevent the basket from moving axially during transport.

The NUHOMS®-61BTH DSC is designed with six alternate basket configurations based on the boron content in the poison plates as listed in Table A.1.4.8-4 or Table A.1.4.8-5 (designated as "A" for the poison plates with the lowest B10 loading to "F" for the highest B10 loading). Three alternate poison materials are allowed: (a) Borated Aluminum alloy, (b) Boron Carbide/Aluminum Metal Matrix Composite (MMC), or (c) Boral®. The poison plates provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

A.1.4.8.3 NUHOMS®-61BTH DSC Contents

Each of the NUHOMS®-61BTH DSC Type 1 and Type 2 configurations is designed to transport intact (including reconstituted) and/or damaged BWR fuel assemblies as specified in Table A.1.4.8-2 and Table A.1.4.8-3. In addition, the 61BTHF can transport up to four failed fuel assemblies placed in Failed Fuel Cans as described in Table A.1.4.8-2. The fuel to be transported is limited to a maximum lattice average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable fuel assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time is 7 years.

Reconstituted fuel assemblies containing up to *four* replacement irradiated stainless steel rods per assembly or 61 lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods are acceptable for storage in 61BTH DSCs as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel rods or 61 with UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

The NUHOMS®-61BTH DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the 2x2 compartments located at the outer edge of the DSC as shown in Figure A.1.4.8-9. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or

Table A.1.4.8-2
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-61BTH DSC

PHYSICAL PARAMETERS:	
Fuel Class	Intact or damaged or failed 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.8-3. Damaged fuel assemblies beyond the definition contained below are not authorized for transport in damaged fuel locations shown in Figure A.1.4.8-9.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel rods is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Failed Fuel	Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly can not be handled by normal means. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its content shall be less than 705 lbs.
RECONSTITUTED FUEL ASSEMBLIES:	
• Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods	4
• Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly	*4*
• Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	61
No. of Intact Assemblies	≤61

Table A.1.4.8-2
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-61BTH DSC
(concluded)

No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 61BTH DSC. Damaged fuel assemblies may only be transported in the 2x2 compartments as shown in Figure A.1.4.8-9. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
No. and Location of Failed Assemblies	Up to 4 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Failed fuel assemblies are to be placed as shown in Figure A.1.4.8-9. Failed fuel assembly/fuel debris is to be encapsulated in an individual Failed Fuel Can (FFC) provided with a welded bottom closure and a removable top closure.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS(1): Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1.4.8-4 or Table A.1.4.8-5.
Fuel Assembly Average Burnup and minimum Cooling Time	Type 1 Per Table A.1.4.8-6.
	Type 2 Per Table A.1.4.8-7.
Decay Heat per DSC	≤22.0 kW for Type 1 DSC, per Figures A.1.4.8-1 through A.1.4.8-4
	≤24.0 kW for Type 2 DSC, per Figures A.1.4.8-1 through A.1.4.8-8
Minimum B10 Content in Poison Plates	Per Table A.1.4.8-4 or Table A.1.4.8-5.

Notes:

(1) Minimum cooling time is the longer of that given in Table *A*.1.4.8-6/*A*.1.4.8-7 and that calculated via the decay heat equation based on the restrictions provided in Figures A.1.4.8-1 thru A.1.4.8-8.

Table A.1.4.8-3
BWR Fuel Assembly Design Characteristics[1] for the NUHOMS®-61BTH DSC

Transnuclear ID	Initial Design or Reload Fuel Designation	Max Length (in) (Unirradiated)	Fissile Material	Maximum No. of Fuel Rods	*Maximum Initial Uranium Content (kg)*
7x7-49/0	GE1 GE2 GE3	176.6	UO_2	49	*198*
8x8-63/1	GE4	176.6	UO_2	63	*192*
8x8-62/2	GE-5 GE-Pres GE-Barrier GE8 Type I	176.6	UO_2	62	*192*
8x8-60/4	GE8 Type II	176.6	UO_2	60	*192*
8x8-60/1	GE9 GE10	176.6	UO_2	60	*192*
9x9-74/2	GE11 GE13	176.6	UO_2	74	*192*
10x10-92/2	GE12 GE14	176.6	UO_2	92	*192*
7x7-49/0	ENC-IIIA	176.6	UO_2	49	*198*
7x7-48/1Z	ENC-III[2]	176.6	UO_2	48	*198*
8x8-60/4Z	ENC Va ENC Vb	176.6	UO_2	60	*192*
8x8-62/2	FANP 8x8-2	176.6	UO_2	62	*192*
FANP 9x9	FANP9 9x9[3]	176.6	UO_2	81	*192*
Siemens QFA	9x9	176.6	UO_2	72	*192*
10x10-91/1	ATRIUM 10, ATRIUM 10XM	176.6	UO_2	91	*192*
ABB-8x8	SVEA-64	176.6	UO_2	64	*192*
ABB-10x10	SVEA-100[4]	176.6	UO_2	100	*192*
LaCrosse	Allis Chalmers-10x10 Exxon/ANF 10x10	125	UO_2	100	*125*

(1) Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.
(2) Includes ENC-IIIE and ENC-IIIF.
(3) Includes FANP 9x9-72, 9x9-79, 9x9-80, and 9x9-81.
(4) Includes SVEA-92, SVEA-96, SVEA-96+, SVEA-96 OPTIMA, SVEA-96 OPTIMA 2.

Table A.1.4.9-1
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-69BTH DSC

PHYSICAL PARAMETERS:	
Fuel Class	Intact or damaged 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.9-2. Damaged fuel assemblies beyond the definition contained below are not authorized for transport.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	4 *4* 69
No. of Intact Assemblies	≤69
No. and Location of Damaged Assemblies	Up to 24 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 69BTH DSC. Damaged fuel assemblies may only be transported in the four outer "6-compartment" arrays as shown in Figure A.1.4.9-1. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS:	
Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1.4.9-3.
Allowable Heat Load Zoning Configurations for each 69BTH DSC	Per Figure A.1.4.9-2 or Figure A.1.4.9-3 or Figure A.1.4.9-4 or Figure A.1.4.9-5.
Fuel Assembly Average Burnup and minimum Cooling Time	Per Table A.1.4.9-4
Decay Heat per DSC	Per Figure A.1.4.9-2 or Figure A.1.4.9-3 or Figure A.1.4.9-4 or Figure A.1.4.9-5.
Minimum B10 Content in Poison Plates	Per Table A.1.4.9-3.

Table A.1.4.9-2
BWR Fuel Assembly Design Characteristics[1] for the NUHOMS®-69BTH DSC

Transnuclear ID	Initial Design or Reload Fuel Designation	Max Length (in) (Unirradiated)	Fissile Material	Maximum No. of Fuel Rods	*Maximum Initial Uranium Content (kg)*
7x7-49/0	GE1 GE2 GE3	176.6	UO_2	49	*198*
8x8-63/1	GE4	176.6	UO_2	63	*192*
8x8-62/2	GE-5 GE-Pres GE-Barrier GE8 Type I	176.6	UO_2	62	*192*
8x8-60/4	GE8 Type II	176.6	UO_2	60	*192*
8x8-60/1	GE9 GE10	176.6	UO_2	60	*192*
9x9-74/2	GE11 GE13	176.6	UO_2	74	*192*
10x10-92/2	GE12 GE14	176.6	UO_2	92	*192*
7x7-49/0	ENC-IIIA	176.6	UO_2	49	*198*
7x7-48/1Z	ENC-III[2]	176.6	UO_2	48	*198*
8x8-60/4Z	ENC Va ENC Vb	176.6	UO_2	60	*192*
8x8-62/2	FANP 8x8-2	176.6	UO_2	62	*192*
FANP 9x9	FANP9 9x9[3]	176.6	UO_2	81	*192*
Siemens QFA	9x9	176.6	UO_2	72	*192*
10x10-91/1	ATRIUM 10, ATRIUM 10XM	176.6	UO_2	91	*192*
ABB-8x8	SVEA-64	176.6	UO_2	64	*192*
ABB-10x10	SVEA-100[4]	176.6	UO_2	100	*192*
LaCrosse	Allis Chalmers-10x10 Exxon/ANF 10x10	125	UO_2	100	*125*

(1) Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.
(2) Includes ENC-IIIE and ENC-IIIF.
(3) Includes FANP 9.9-72, 9x9-79, 9x9-80, and 9x9-81.
(4) Includes SVEA-92, SVEA-96, SVEA-96+, SVEA-96 OPTIMA, SVEA-96 OPTIMA 2.

Proprietary Information Withheld from Pages A.2.13.11-i, -ii, and A.2.13.11-1 through A.2.13.11-11 Pursuant to 10 CFR 2.390

Proprietary Information Withheld Pursuant to 10 CFR 2.390

A.2.13.11.4 *Deleted*

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Proprietary Information Withheld Pursuant to 10 CFR 2.390

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Proprietary Information Withheld from Pages A.2.13.11-23 through A.2.13.11-53
Pursuant to 10 CFR 2.390

Figures A.2.13.11-16 through -26 on pages A.2.13.11-54 through -64 have been removed.

Chapter A.3
Thermal Evaluation

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The decay heat generation rates used in the transient model are listed below.

Decay Heat Generation Rate

DSC Type	Heat Load (kW)	Heat Load (Btu/hr)	D_i (in)	L_b (in)	Decay heat Generation Rate (Btu/hr-in^3)
69BTH	26.0	88,720	68.75	164	0.1457
	32.0	109,194	68.75	164	0.1794
24PTH	26.0	88,720	66.19	168.60	0.1529

All the assumptions and conservatism described in Section A.3.3.1.1 for the MP197HB model are valid for determination of initial conditions.

A.3.4.2 Fire Test Conditions

No fire test is performed. Instead, the fire conditions are simulated using the finite element model of the MP197HB TC.

Based on the requirements in 10 CFR 71, part 73 [6], a fire temperature of 1475 °F, fire emissivity of 0.9 and a period of 30 minutes are considered for the fire conditions. A bounding forced convection coefficient of 4.5 Btu/hr-ft^2-°F is considered during burning period based on data from reference [13]. Surface emissivity of 0.8 is considered for the packaging surfaces exposed to fire based on 10 CFR 71, part 73 [6].

The total heat transfer coefficient during fire is determined using the following equations.

$$h_{t,fire} = h_{r,fire} + h_{c,fire}$$

Where,

$h_{r,fire}$ = fire radiation heat transfer coefficient (Btu/hr-in^2-°F)

$h_{c,fire}$ = forced convection heat transfer coefficient during fire = 4.5 Btu/hr-in^2-°F

The radiation heat transfer coefficient, $h_{r,fire}$, is given by the equation:

$$h_{r,fire} = \varepsilon_w F_{wf} \left[\frac{\sigma(\varepsilon_f T_f^4 - T_w^4)}{T_f - T_w} \right] \quad \text{Btu/hr-in}^2\text{-°F}$$

where,

ε_w = TC outer surface emissivity = 0.8 [6]

ε_f = fire emissivity = 0.9 [6]*

F_{wf} = view factor from TC surface to fire = 1.0

σ = 0.1714 $\times 10^{-8}$ Btu/hr-ft^2-°R^4

T_w = surface temperature (°R)

T_f = fire temperature = 1475°F = 1,935°R

The sensitivity study that documents the effects of fire emissivity of 1.0 on the thermal performance of the MP197HB TC is documented in Appendix A.3.6.8.

The following gaps are reduced from 0.0625" under NCT to 0.01" under HAC to maximize the heat input from the fire toward the cask after free drop:

a) 0.01" axial gap between thermal shield and impact limiter case
b) 0.01" axial gap between thermal shields and cask top or bottom end surface

$$n_{max} = \frac{R_{j,model}}{R_{j,Al-Al}} - 1$$

$$n_{max} = \frac{2.486E-3}{4.164E-4} - 1 = 4.97$$

This shows that at least four plates, three aluminum plates and one poison plate can be paired together in 69BTH basket without affecting the thermal performance evaluated in this calculation.

A.3.6.8 Sensitivity Study for Effects of Fire Emissivity

A fire emissivity (ε_f) of 0.9 was considered in Section A.3.4.2 to calculate the fire radiation heat transfer coefficient ($h_{r,fire}$). Assuming conservatively, the fire as a black body, an emissivity of 1.0 can be considered for the fire. The effect of this assumption is enveloped for the MP197HB TC in a sensitivity analysis in this section considering the maximum heat load of 32 kW with the external fins installed on the shield shell. The only change is the increase of the fire emissivity (ε_f) from 0.9 to 1.0 in the input file for running the finned TC under HAC. The maximum component temperatures from the sensitivity run with ε_f=1.0 are shown in the following table.

Maximum Component Temperatures for ε_f = 1.0

DSC type	69BTH DSC, with 32 kW heat load, Finned MP197HB TC, ε_f = 1.0			
Component	**Time (hr)**	**T_{max} (°F)**	**T_∞ (°F)(2)**	**Limit (°F)**
DSC shell	7.0	512	(3)	---
DSC shell at mid-length (1)	7.0	512	(3)	---
Cask inner shell	1.9	497	400	---
Gamma shield	0.5	571	399	621 [5]
Outer shell	0.5	720	382	---
Shield shell	0.5	1440	335	---
Cask lid	13.0	315	309	---
Cask bottom plate	1.0	416	383	---
Cask lid seal	10.0	323	314	400 [18, 19]
Vent & test seal at top	13.0	313	308	400 [18, 19]
Ram plate seal	1.9	380	377	400 [18, 19]
Test seal at bottom	13.0	382	377	400 [18, 19]
Drain port seal at bottom	10.0	388	381	400 [18, 19]
Helium in TC cavity	4.0	389	380	---

Notes:
(1) This value is the maximum DSC shell temperature in the region where the fuel assemblies have the maximum peaking factor.
(2) These values are retrieved from the transient model at 27.0 hrs after the end of the fire accident. Based on the time-temperature histories for the original TC model shown in Figure A.3-40 through Figure A.3-42, the steady state temperatures are bounded by these temperatures.
(3) Due to the adiabatic boundary conditions considered conservatively for the steady state cool-down runs (described in Section A.3.4.2), the maximum DSC shell temperature at the end of the transient run remains bounded by the steady state temperature of 537°F reported for the 69BTH DSC with 32kW heat load in Table A.3-17.

Since the fire emissivity does not have any effect on the steady state temperatures after cool-down, the maximum DSC shell temperature for the 69BTH with 32 kW heat load remains bounded by 537°F as reported in Table A.3-17.

A comparison of the maximum TC component temperatures for ε_f of 1.0 to those for ε_f of 0.9 are shown in the following table.

Comparison of the Maximum TC Component Temperatures for ε_f of 1.0 and 0.9

	Transient Temperature			Steady State Temperature		
	ε_f =1.0	ε_f = 0.9		ε_f =1.0	ε_f = 0.9	
Component	T_{max} (°F)	T_{max} (°F)	ΔT_{max} (°F)	T_∞ (°F)(2)	T_∞ (°F)(3)	ΔT_∞ (°F)
DSC shell	512	509	+3	(4)	(4)	0
DSC shell at mid-length(1)	512	509	+3	---	---	N/A
Cask inner shell	497	487	+10	400	403	-3
Gamma shield	571	552	+21	399	401	-2
Outer shell	720	690	+30	382	383	-1
Shield shell	1440	1393	+47	335	337	-2
Cask lid	315	314	+1	309	311	-2
Cask bottom plate	416	416	0	383	384	-1
Cask lid seal	323	321	+2	314	316	-2
Vent & test seal at top	313	311	+2	308	310	-2
Ram plate seal	380	380	0	377	378	-1
Test seal at bottom	382	380	+2	377	379	-2
Drain port seal at bottom	388	386	+2	381	383	-2
Helium in TC cavity	389	387	+2	380	380	0

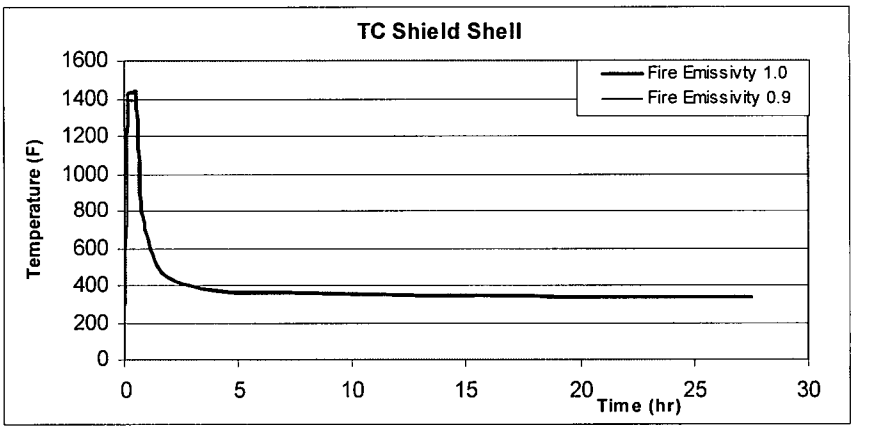
Notes:
(1) This value is the maximum DSC shell temperature in the region where the fuel assemblies have the maximum peaking factor.
(2) These values are retrieved from the transient model at 27.0 hrs after the end of the fire accident.
(3) These values are retrieved from the transient model at 20.0 hrs after the end of the fire accident.
(4) Due to the adiabatic boundary conditions considered conservatively for the steady state cool-down runs (described in Section A.3.4.2), the maximum DSC shell temperature at the end of the transient run remains bounded by the steady state temperature of 537°F reported for the 69BTH DSC with 32kW heat load in Table A.3-17.

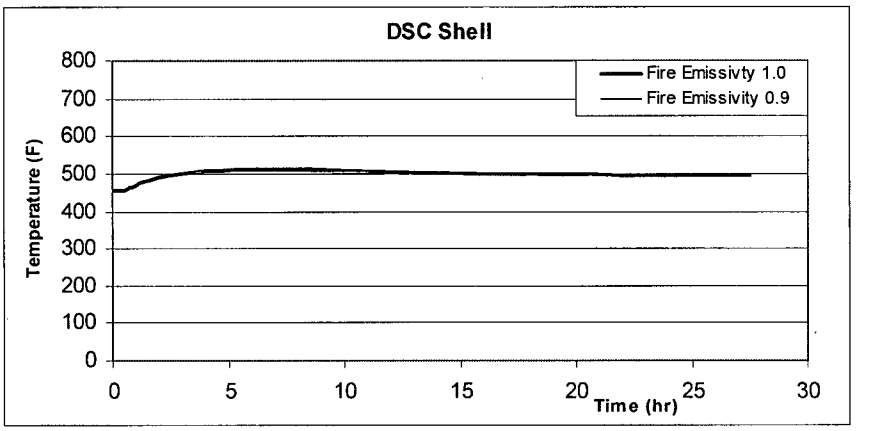
As seen in the above table, the largest effect of increasing the fire emissivity from 0.9 to 1.0 occurs during the short period of the fire at the shield shell, which is directly exposed to the fire. The other components remain shielded from the fire effect so that the cask inner shell temperature increases by only 10°F and the DSC shell temperature increases only by 3°F during the transient run.

These temperature increases are relatively small and occur for a short period of time and therefore do not affect the thermal and structural performance of the MP197HB TC.

The containment seals are protected from direct fire exposure by the impact limiters. The effect of increasing the fire emissivity from 0.9 to 1.0 on the maximum seal temperatures is limited to 2°F for a short period of time after the fire. The transient and the steady state temperatures of the containment seals remain well below the temperature limit of 400°F [18, 19]. Therefore the containment function of the seals remains unaffected by the increase of the fire emissivity from 0.9 to 1.0.

The time temperature histories for the TC shield shell and DSC shell from the sensitivity study with ε_f of 1.0 are compared to those from the original model with ε_f of 0.9 in the following figures, respectively.





Results of the Sensitivity Study

As seen in Figure A.3-40 through Figure A.3-42, all the TC component temperatures decrease through the cool-down period. The small differences seen in the comparison table between the steady state temperatures are caused by the fact that the transient temperatures at different hours (27.0 hours for sensitivity run and 20.0 hours for the original run) are used to bound the steady state temperatures. Since the TC component temperatures are decreasing, the values at 27.0 hours for the sensitivity run are lower than those for the original run. It is expected that the actual steady state temperatures in both runs achieve the same values and are independent of the fire emissivity.

It is evident from the first table in this section that the maximum TC component temperatures remain well below the allowable limits.

In conclusion, the effect of increasing the fire emissivity from 0.9 to 1.0 occurs only for a short period of time on the outermost components of the TC exposed to the fire. The function of these TC components remains unaffected by this change in the fire emmissivity.

Chapter A.5
Shielding Evaluation

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parts nomenclature and materials weights for these assemblies are provided in Table A.5-8 and Table A.5-9.

The design parameters for the various BWR fuel assemblies are provided in Chapter A.2, Appendix A.2.13.11, Table A.2.13.11-2 (rod drop analysis). This information is also provided in Chapter A.6, Appendix A.6.5.1, Table A.6.5.1-3 and Table A.6.5.1-53 and Appendix A.6.5.2, Table A.6.5.2-2 (criticality analysis).

The design parameters for the various PWR fuel assemblies are provided in Chapter A.2, Appendix A.2.13.11, Table A.2.13.11-3 (rod drop analysis). This information is also provided in Chapter A.6, and Appendix A.6.5.4, Table A.6.5.4-4 (criticality analysis).

The design limits for the various fuel assembly designs are provided in Chapter A.1, Appendix A.1.4.1 through A.1.4.9. The maximum initial uranium loading of all the other BWR fuel assembly designs (except the GE 7x7 design) are restricted to 0.193 MTU. Similarly, the maximum initial uranium loading of all the other PWR fuel assembly designs (except the B&W 15x15 design) are restricted to 0.482 MTU. This ensures that the design basis fuel assembly designs (for BWR and PWR fuel assemblies) as discussed earlier, are bounding from a source term calculation standpoint.

Assembly hardware information for all the various fuel assembly designs is not included in the SAR, (except for the design basis fuel assemblies as shown in Table A.5-8 and Table A.5-9) however, the MTU loading is the most important parameter from a source term calculation standpoint. The fuel assembly hardware for the GE 7x7 fuel assembly is bounding as it contains the maximum amount of steel and inconel than any other BWR fuel design. For the PWR fuel assembly designs, (due to the credit for fuel assembly burnup in the criticality evaluation (Chapter A.6) that require cooling times greater than or equal to 15 years) the contribution from the fuel assembly hardware becomes less important due to substantial decay of the Co-60 source from hardware irradiation.

The spent fuel payload consists of various DSCs with BWR fuel assemblies with or without channels and is specified in Chapter A.1, Appendix A.1.4.7 through A.1.4.9. The source term calculations for DSCs with the BWR fuel payload include the contribution from the channel (Table A.5-8) while the shielding calculations do not take credit for them. This represents conservatism in the gamma dose rate calculations by approximately 15% for fuel assemblies with channels (typically the most representative of all loaded BWR fuel assemblies).

The spent fuel payload consists of various DSCs with PWR fuel assemblies and associated control components (CCs) and is specified in Chapter A.1, Appendix A.1.4.1 through A.1.4.6. For the PWR fuel assemblies, the various authorized CCs are listed in the above appendices. These include PWR burnable poison rod assemblies, thimble plug assemblies, control rod assemblies, control rod cluster assemblies, axial power shaping rods, orifice rod assemblies, vibration suppression inserts, neutron source assemblies, and neutron sources. The CCs are typically solid or hollow rods of stainless steel or zircaloy containing neutron absorbing or neutron source materials. Typically, the source term from these CCs is dominated by the Co-60 spectrum. The source term is provided in Table A.5-13 of the SAR. The maximum allowable

source term per DSC is limited to that specified in the above appendices for the DSCs containing PWR fuel assemblies.

Therefore, a separate material composition and irradiation history is not necessary for characterizing all of these CCs. A description of the source term calculation for the CCs is provided in Section A.5.2.2.2. A discussion of the adjustment to the cooling times for qualification of fuel assemblies containing CCs is provided in Section A.5.5.2.

Reconstituted fuel assemblies are those where one or more fuel rods are replaced with "reconstituted" rods that displace the same amount of moderator in the active fuel region. Table A.5-9 of the SAR provides material details of a reconstituted fuel assembly where the rods are replaced with solid stainless steel after one cycle of irradiation. This assembly undergoes two additional cycles of irradiation where the source terms of the original and reconstituted fuel assemblies are compared. The summary of these evaluations is discussed in Section A.5.5.3.

Partial length shielding assemblies (PLSAs) are only authorized for the 24PTH DSC and are Westinghouse 15x15 design fuel assemblies that consist mostly of stainless steel. They are restricted to a maximum burnup of 40 GWD/MTU and a maximum MTU loading of 0.330 as shown on Chapter A.1, Appendix A.1.4.3. Therefore, they are bounded by the design basis B&W 15x15 fuel assemblies.

Fuel qualification tables are established for each DSC to assure compliance with maximum transportation dose rates criteria. Radiological source terms are calculated using SAS2H/ORIGEN-S modules of SCALE [1]. Two depletion models are used for radiological source calculation. They are based on the DB BWR and PWR FA parameters and characteristics. Radiological sources from these two depletion models are used for qualification of BWR and PWR assemblies for transportation, respectively. The model presented in Appendix P, Section P.5.5.1 of reference [18] is used for DB PWR FA. SAS2H\ORIGEN-S depletion model for DB BWR FA is developed using design characteristics and components presented in Table A.5-7 and Table A.5-8.

To perform fuel qualification for the transportation in the cask, response functions for all the transportation cask\DSC shielding configurations are established. The response functions were also used for determination of the bounding cask\DSC shielding configuration for hypothetical accident conditions (HAC). Details on the fuel qualification for transportation purpose and calculation of response functions are provided in Section A.5.5.1.

The cask shielding performance is based on a bounding shielding evaluation. The MP197HB cask loaded with 69BTH DSC containing DB BWR assemblies results in bounding dose rates. Because of that, DB BWR assembly is simply referred to as DB FA for the shielding evaluation in the text of this chapter. The DB FA for normal conditions of transport (NCT) dose rate analysis has an initial enrichment of 3.8 wt. % U-235 bundle-average burnup of 55,000 MWD/MTU with a 7 3/4 year decay time. The DB FA with an enrichment of 4.3 wt. % U-235 and a bundled-average burnup of 70,000 MWD/MTU and 21.0 year decay time generates radiological sources for the shielding performance evaluation of the cask at HAC.

NCT configurations are modeled with the neutron shield and impact limiters intact. This shielding calculation is performed using the Monte Carlo computer code MCNP [5]. Dose rates on the side, top and bottom of the MP197HB package are calculated for the various sources described in Section A.5.2 and summed to give a total gamma and neutron dose rate.

HAC shielding evaluation assumes that 75% of the neutron shield is lost. The impact limiters are assumed to be crushed 12" axially and the wood is removed. In addition, the top and bottom 0.375 inches of lead (axial direction) is removed to account for lead slump. Finally, the lead gamma shield radial thickness is reduced by 0.1". These assumptions result in a more severe degradation of the cask shielding properties than the accident conditions shown in Chapter A.2. Tests have shown that the neutron shielding material retains more than 60% of its principal contents (hydrogen, boron) following a design basis fire accident and a 25% credit employed in the shielding calculations is conservative. Shielding calculations for the HAC are also performed using the MCNP code.

Fuel qualification tables (FQTs) are established for each DSC type and authorized loading configuration. The applicable FQTs for each DSC are presented in Chapter A.1, Appendix A.1.4.1.1 through Appendix A.1.4.1.9. The individual FQTs for each DSC/loading configuration span the authorized burnup and initial enrichment combinations allowed for the fuel to be transported. The transportation FQTs are determined to provide minimum required cooling time to comply only with the dose rates limit criteria for transportation. It is important to remember however that the minimum cooling times provided in these FQTs may not be bounding for criteria other than shielding. These tables only provide the minimum required cooling time to meet the transportation dose rate limits. Other considerations are decay heat limits and burnup credit limits depending on the specific DSC and loading configuration.

To evaluate the decay heat for a given assembly, one must use the applicable decay heat equations (DHE) provided in Section A.5.5.4. Two sets of DHEs are established; one set for BWR (maximum of 0.198 MTU) fuel and one set for PWR (maximum of 0.490 MTU) fuel.

The expected dose rates (for NCT and HAC) from the MP197HB package with a DSC loaded with spent fuel are summarized in Tables A.5-1 through A.5-3. Maximum dose rates at various distances from the cask are provided in Tables A.5-21 through A.5-26. The spatial distributions of the dose rates at various distances from the cask are shown on plots of Figure A.5-10 through Figure A.5-15. The results show that the MP197HB transportation package complies with dose rates restrictions of 10 CFR 71 during NCT and HAC.

Two sets of package surface dose rates are shown in Table A.5-1. The first set under the column "Transport Package Surface" provides maximum dose rates at the external surface of the package. The surfaces where the maximum dose rates are calculated in the axial direction (top and bottom ends) are located at the outer surface of the impact limiters. The surfaces where the maximum dose rates are calculated in the radial direction (side) are located at the outer surface of the cask. Table A.5-23 does not include the surface dose rate since this is very conservative and is not an accessible surface during NCT. The second set under the column "Vehicle Edge" provides the maximum dose rates at the external surface of the package in the transport configuration. All the maximum dose rates in the radial and axial direction are calculated at the outer surface of the impact limiters. Note 1 of the Table A.5-1 provides the appropriate clarification.

The radial dose rates for these two sets of dose rates are different because they are calculated at different locations while the axial dose rates are identical.

Dose rates restriction criteria for the transportation are the following:

- External dose rate at any point on the outer accessible surface of the vehicle under normal conditions: 200 mrem/hr (max).
- External dose rate at any point 2 m from the outer lateral surfaces of the vehicle under normal conditions: 10 mrem/hr (max).
- External dose rate at any point 1 m from the outer lateral surfaces of the vehicle under hypothetical accident conditions: 1000 mrem/hr (max).
- The transportation package must not result in dose rates greater than 2.0 mrem/hr at occupied locations near the package during the transportation. Ends of the conveyance are considered the occupied locations.

A.5.1.2 Irradiated/Contaminated Waste

The NUHOMS®-MP197HB is designed for shipment of various types of irradiated and contaminated reactor hardware. The payload will vary from shipment to shipment and will consist predominately of components specified in a numbered list of Section A.5.2. The NUHOMS®-MP197HB is designed to transport a payload of 55.0 tons of dry irradiated and/or contaminated non-fuel bearing solid materials in secondary containers. The safety analysis of the cask takes no credit for the containment provided by secondary containers.

The quantity of radioactive material is limited to a maximum of 8,182 A_2 (90,000 Ci of Co60). The radioactive material is primarily in the form of neutron activated metals, or metal oxides in solid form. Surface contamination may also be present on the irradiated components. When a wet load procedure (i.e., in-pool) is followed for cask loading, the cask cavity and secondary container are drained and dried to ensure that free liquids do not remain in the package during transport.

All these materials are basically neutron activated hardware that are mostly composed of steel and are defined by a Co-60 source spectrum. Therefore, detailed material composition and irradiation history is not necessary for the characterization of this waste. As discussed above, since the maximum quantity of the radioactive material is restricted to 90,000 Ci of Co-60 (equivalent) and the total payload mass is restricted to 55 tons, the shielding calculations are performed using this maximum allowable source terms. A list of the various types of irradiated hardware materials (not necessarily an exhaustive list) is provided in Section A.5.2. A discussion of the source specification based on the maximum allowable limits is provided in Section A.5.2.2.3.

The decay heat load of the radioactive material is expected to be less than 5 kW, which is well below the 26 kW limit for the cask.

NCT and HAC dose rates from the transportation cask containing the irradiated waste canister are summarized in Table A.5-1, Table A.5-2, Table A.5-5 and also Table A.5-27, Table A.5-28.

Dose rate distributions at various distances from the cask are plotted on Figure 5-14 and Figure 5-15

A.5.2 Source Specification

There are five principal sources of radiation associated with transport of spent nuclear fuel that are of concern for radiation protection:

- Primary gamma radiation from spent fuel,
- Primary neutron radiation from spent fuel (both alpha-n reactions and spontaneous fission),
- Gamma radiation from activated fuel structural materials and fuel inserts,
- Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask, and
- Neutrons produced by sub-critical multiplication in the fuel.

The MP197HB package is designed to transport LWR class of fuel assemblies in various DSC designs or dry irradiated and/or contaminated non-fuel bearing solid material contained in a secondary container. The fuel assemblies as a function of DSC type are given in Chapter A.1, Appendix A.1.4.1.1 through Appendix A.1.4.1.9. The various fuel assembly designs were separated according to fuel assembly array, the maximum metric tons of uranium, and the number of guide /instrument tubes. These parameters are the significant contributors to the SAS2H/ORIGEN-S model. The largest uranium loading results in the largest source term at any chosen enrichment and burnup, thus the B&W 15x15 Mark B10 and GE-2,3 7x7 Type G2A are bounding assembly types for PWR and BWR spent fuel assemblies, respectively. It was established that the cask loaded with 69BTH DSC containing bounding BWR assemblies results in the bounding dose rates.

Table A.5-7 and Table A.5-8 provide characteristics and components of the design basis BWR fuel assembly. The SAS2H/ORIGEN-S modules of the SCALE code are used to generate gamma and neutron source terms for the design basis fuel assembly. The design basis radiological sources for NCT and HAC are due to DB FA irradiated at a constant specific power of 12.4 and 15.8 MW/assembly to a total bundle average burnup of 55,000 and 70,000 MWD/MTU, respectively. A three-cycle operating history is utilized. The assembly is burned for 292 effective full power days per cycle, for the duration of 3 cycles with 72 day down time after each cycle except for no down time in the last cycle.

The source terms used in the bounding shielding evaluation are generated for the fuel assembly active fuel region, the plenum region, and the top and bottom end fitting regions. The fuel assembly hardware materials and masses on a per assembly basis are listed in Table A.5-8. Table A.5-10 provides the material composition of fuel assembly hardware materials. Cobalt impurities are included in the SAS2H model.

The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2 and 0.15, respectively [4] in the BWR FA model. These factors are used to correct for the spatial and spectral changes of the neutron flux outside of the active fuel zone. The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis.

Axial variation in the moderator density along the BWR fuel assembly was considered by including a volume averaged density for the moderator around the fuel pins. Axial variation of temperatures and moderator densities from reference [1] were used to calculate the volume average moderator density for use in the BWR source term models. The fuel assembly water temperature and volume averaged density used are 558 K and 0.4323 g/cm^3, respectively. The fuel channel water temperature and density are 552 K and 0.669 g/ cm^3, respectively. Because temperature of moderator varies significantly in BWR fuel assembly along axial direction, different moderator densities are used in SAS2H\ORIGEN-S models when calculating radiological sources due to different axial exposure regions. The water is coolest near bottom of the fuel assembly and its temperature rises until it reaches the top of the fuel assembly. One can assume that water density is nearly the same inside and outside of the fuel channel at the bottom region of the FA. A water density of 0.743 g/cm^3 is used for the bottom region and a density of 0.264 g/cm^3 is used inside of the fuel channel near the plenum/top nozzle area. Channel water temperature at the bottom and top regions is 552 K and 558 K, respectively.

The *44GROUPNDF5* cross section library was used during SAS2H/ORIGEN-S depletion calculations. This library contains data on approximately 300 nuclides from the ENDF/B-V data. It was designed to accommodate two windows in oxygen cross section spectrum, a window in the iron cross section, the Maxwellian peak in the thermal range and the 0.3-eV resonance in ^{239}Pu.

Applicability of SAS2H for prediction of isotopic content in BWR assemblies was analyzed in [13]. A UO_2 sample was burned to 57 GWd/MTU in a BWR reactor. The sample U-235 enrichment was 4.97 wt. %. Also, the isotopic content of the discharged sample was measured experimentally. Measured content was reported for actinides and fission products. Among concentrations of 16 nuclides investigated, 5 agreed with the measured values to within ±5%. Nine predictions were within ±10%. Also sensitivity to burnup for isotope concentrations was investigated as burnup varied between 90 and 110%. The results suggest that there is no reason to expect that the ongoing evaluations of the higher burnup fuel will result in less favorable comparisons. Therefore, the uncertainty in the gamma source term, and associated dose rates, is estimated to be within ±10%.

Additional evaluations of the existing Light Water Reactor (LWR) fuel isotopic data with SAS2H and the 44-group ENDF/B-V library used in the calculation of the design basis source terms are documented in References [16] and [17]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [14]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. There are ongoing efforts, some of which are documented in Reference [12], to obtain more data for burnups above 45 GWd/MTU.

As noted in References [15] and [12], there is no public data for the neutron component currently available that bounds a fuel burnup of up to 70 GWd/MTU. However, as documented in Reference [15] and confirmed in the SAS2H analysis, the total neutron source with increasing

burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of ^{244}Cm (more than 90% of all neutrons both spontaneous fission and (α,n)) with (α, O–18) of lesser importance. Reference [13] documents that SAS2H tends slightly to over predict the concentration of ^{244}Cm when burnup is varied during the sensitivity study. Therefore, as the ^{244}Cm isotope accounts for more than 90% of the total neutron source term, the uncertainty in the neutron source and associated neutron dose rates is expected to be less than ±11%.

As documented in Reference [15] and as observed in preparing the fuel qualification tables, the gamma dose rate increases nearly linearly with burnup relative to the direct gamma component and the neutron dose rate increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWd/MTU, the contribution from neutron (and associated n, γ) components to the total dose rates measured on the surfaces of the cask increases in relative importance to that of the gamma component.

As discussed above, any impact of uncertainties in source terms is not expected to be significantly greater than 10 %, for the transportation system. Therefore, depletion calculations with SAS2H for calculation of some terms are appropriate for fuel burned above 45 GWd/MTU.

The uncertainty value of 10% is an uncertainty in the ability of the SAS2H code to predict the isotopic concentration of nuclides in the fuel. In many cases, this results in SAS2H over-predicting the quantity of certain fission product or actinide isotope, thereby resulting in a conservative prediction of source terms. These benchmarks demonstrate that the neutron spectrum calculated by SAS2H during depletion is appropriate for the purpose of source term calculations. An uncertainty has not been applied in the dose rate calculations.

The NUHOMS®-MP197HB is also designed for shipment of various types of irradiated and contaminated reactor hardware. The payload will vary from shipment to shipment and will consist predominately of the following components either individually or in combinations:

1. BWR Control Rod Blades
2. BWR Local Power Range Monitors (LPRMs)
3. BWR Fuel Channels
4. BWR Poison Curtains
5. PWR Burnable Poison Rod Assemblies
6. Reactor Vessel and Internals (PWR and BWR)

The typical Cobalt 60 specific activity ranges for these items are as follows:

1. Control Rod Blades	$1.3\times10^{-4} - 1.1\times10^{-2}$ Ci/g
2. LPRMs	$1.0\times10^{-2} - 4.8\times10^{-2}$ Ci/g
3. Fuel Channels	$7.8\times10^{-5} - 2.0\times10^{-4}$ Ci/g
4. Poison Curtains	$6.2\times10^{-4} - 4.0\times10^{-2}$ Ci/g

An evaluation of the contribution from neutron radiation source to the total dose rate was performed using response functions. It showed that the neutron radiation dose rates for MP197HB cask loaded with 69BTH DSC containing neutron radiation source from Table A.5-12 (the one that was subsequently used for HAC shielding evaluation) results in bounding neutron radiation dose rates. The cask configuration for HAC results in the dose rates that are nearly totally dominated by neutron radiation source at high burnups. Therefore it was concluded that radiological source from Table A.5-12 will result in bounding HAC dose rates.

When other DSCs are loaded in the cask, presented dose rates at various distances from ends of impact limiters may be slightly different. This is due to differences in the DSCs shield plug thicknesses and source terms distribution and strength in axial exposure regions. However the evaluation presented herein bounds the other cases.

The dose rate distribution on side of the cask body displayed on Figure A.5-10 demonstrates variation of the shielding properties on side of the cask. The distribution can be mapped on surface of the cask using a cylindrical coordinate system with an axis coincident with the DSC\cask axis. Polar (angular) coordinate is measured in rotations from an imaginary plane through sheer key and DSC\cask axis. The center of the shear key is at 0.0 or 1.0 rotations angular coordinate in that system. Centers of the trunnions are at 0.25 and 0.75 rotations. Dose rate variation can be understood if looking on modeled geometry of the cask shown on sketch of Figure A.5-1. It can be seen from the figure that active fuel region on bottom of the cask extends axially beyond the neutron shield on side of the cask. Also the in-core region faces trunnion attachment blocks on bottom portion of the cask side (not shown on Figure A.5-1). The attachment blocks cut large pieces of the neutron shielding material out side of the cask. Because they are made from steel they provide poor protection from neutron radiation. These two geometry features cause peak 1 and 2 on Figure A.5-10. Naturally, if there were no trunnion attachment blocks and neutron shielding would extend throughout an entire length of the active region, dose rate on side of the cask would be highest at the mid of the in-core region, which is marked with 3 on the figure. Solid aluminum transition rails around peripheral DSC fuel compartments provide fair additional shielding, especially against gamma radiation. If looking on Figure A.5-1 one can notice that Top Nozzle region extends above solid aluminum transition rails in the MCNP model for the bounding shielding evaluation. This causes a fourth peak on Figure A.5-10.

NCT maximum dose rates are summarized in Table A.5-1. It presents dose rates on surface of the transport package, edge of the package and 2 meters from the edge. For the HAC, Table A.5-2 summarizes the maximum calculated doses at 1 m from the cask body shield shell.

The dose rates for an individual at the end of the rail car are presented in Table A.5-3. These results are presented as a function of the length of the rail car. *Comparing these dose rates to the maximum dose rates in Table A.5-21 and Table A.5-22, a minimum distance of 3.3 m from the outer surface of the impact limiter is required to ensure that the dose rate at the occupied locations is below the limit of 2.0 mrem/hour.*

The dose rate analysis was performed using MCNP5 [5]. Selected inputs for MCNP are included in Section A.5.7.

Summary on dose rates from the bounding evaluation in Table A.5-1 and Table A.5-2 demonstrates that the package complies with the criteria of transportation specified at the end of Section A.5.1.

A.5.4.2 Irradiated/Contaminated Waste

NCT and HAC dose rates from the transportation cask containing irradiation/contamination waste canister are summarized in Table A.5-1, Table A.5-2, Table A.5-5 and also Table A.5-27, Table A.5-28. Dose rates at various distances from ends of impact limiters of the cask containing irradiated waste canister are shown in Table A.5-27. They are bounding for the dose rates at the similar locations from the cask containing DSCs.

The dose rates for an individual at the end of the rail car are presented in Table A.5-5. These results are presented as a function of the length of the rail car. Comparing these dose rates to the maximum dose rates in Table A.5-27, a minimum distance of 5.5 m from the outer surface of the impact limiter is required to ensure that the dose rate at the occupied locations is below the limit of 2.0 mrem/hour.

Terminology used in the discussion in the previous subsection is applicable to the current subsection too. Dose rate distributions at various distances from the cask are plotted on Figure A.5-14 and Figure A.5-15 in a cylindrical coordinate system with an axis coincident with the DSC\cask axis.

The distribution on Figure A.5-15 can be mapped on side surface of the cask using a cylindrical coordinate system in the same manner as described in the previous section. Dose rate distribution along side of the cask containing the irradiated waste canister has "holes" at certain angular coordinates. Those "holes" are especially pronounced on the figure showing the dose rate distribution at a radial distance corresponding to side of impact limiters. Location of the "holes" coincides with the locations of the trunnions and the shear key. This is because of more steel from the trunnions attachment blocks and shear key at those locations, which provides better shielding against gamma radiation than the neutron shielding material, VYAL-B.

- No additional cooling time is required for assemblies with CCs in 24PT4 DSC because the transportation FQT already accounts for presence of 24PT4 design basis CC sources.
- A matrix shown in Table A.5-14 provides additional cooling times at selected cooling times of CCs. The additional times shown are bounding even when all the fuel outer peripheral compartments contain CCs.
- *For PWR fuel assemblies containing CCs, no additional cooling time is needed when they are loaded with cooling times greater than or equal to 15 years.*

Note: the guidelines are based **only** on shielding performance evaluation.

A.5.5.3 Transportation FQTs for Assemblies with Damaged or Reconstituted Rods

The following guidelines should be applied when fuel assemblies with reconstituted rods are considered for transportation in *the* MP197HB Transportation Cask:

- The maximum number of reconstituted fuel assemblies that can be loaded per DSC is equal to number of assemblies per DSC.
- *PWR* fuel assemblies may contain up to 10 rods *and BWR fuel assemblies may contain up to four rods* that are reconstituted with stainless steel that is irradiated.
- There is no limit on a number of rods reconstituted with un-irradiated stainless steel or zircaloy or low enriched (lower than of an original, un-reconstituted, FA), natural uranium, UO_2 or other non-fuel material.
- There is no effect on the source terms due to the position of the reconstituted rods in the fuel rod array.
- Additional cooling time is required for assemblies in outer peripheral fuel compartments only. The outer peripheral fuel compartments are depicted in the fourth bullet at the end of Section A.5.5.2.
- For cooling times greater than or equal to 10 years no extra cooling is required and the FQT cooling times for un-reconstituted FAs are also applicable for reconstituted FAs.
- One more year of cooling time is required for reconstituted fuel assemblies cooled for less than 10 years and located in outer peripheral DSCs' fuel compartments as depicted in the fourth bullet at the end of Section A.5.5.2.
- *The maximum number of fuel assemblies with irradiated stainless steel reconstituted rods is restricted to four (Chapter A.1, Appendix A.1.4.1 through A.1.4.9) thereby ensuring that the analysis is conservative.*

Note: the guidelines are based **only** on shielding performance evaluation.

A.5.5.4 Decay Heat Restrictions

The various DCS payloads allowed for transportation in the MP197HB cask have individual decay heat load limitations. There are numerous DSC loading options to meet those restrictions.

Table A.5-3
Summary on MP197HB Averaged Dose Rates at Occupied Locations

(Exclusive Use Package for Transportation)

Rail CarLength (feet)	Averaged Dose Rates as a Function of Railcar Length (mrem/hr) (1)								
	Neutron		(n,γ) Gamma		Gamma		Total		Limit
	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom	
40	0.3	1.2	0.1	0.4	0.2	0.4	0.6	2.0	2.0
50	0.2	0.7	0.05	0.2	0.1	0.3	0.4	1.2	
60	0.15	0.5	0.03	0.1	0.08	0.2	0.3	0.8	

(1) Dose rates are averaged over area encompassing 12.8' Ø circle centered at the Cask axis. *Comparing these dose rates to the maximum dose rates in Table A.5-21 and Table A.5-22, a minimum distance of 3.3 m from the outer surface of the impact limiter is required to ensure that the dose rate at the occupied locations is below the limit of 2.0 mrem/hour.*

Table A.5-5
Summary of Average Dose Rates at Occupied Locations Near MP197HB

Containing the irradiated waste canister (Exclusive Use Package for Transportation)

Rail Car Length (feet)	**Averaged(1)**		**Limit**
	Top	**Bottom**	
40	3.3	3.3	2.0
50	2.1	2.1	
60	1.4	1.4	

(1) Dose rates are averaged over area encompassing 12.8' Ø circle centered at TC canister axis. *Comparing these dose rates to the maximum dose rates in Table A.5-27, a minimum distance of 5.5 m from the outer surface of the impact limiter is required to ensure that the dose rate at the occupied locations is below the limit of 2.0 mrem/hour.*

package may be shipped by a carrier in a nonexclusive conveyance, from a criticality requirements point of view.

A.6.1.3 Summary of Criticality Evaluations

A brief summary of the criticality analysis results for each of the payloads along with the source are presented in the following table:

Payload	K_{EFF}	USL	Reference Appendix
NUHOMS®-24PT4 DSC	0.9393	0.9411	A.6.5.3
NUHOMS®-32PT DSC	0.9250	0.937*0*	A.6.5.6
NUHOMS®-24PTH DSC NUHOMS®-24PTHF DSC	0.9250	0.937*0*	A.6.5.5
NUHOMS®-32PTH and 32PTH Type 1 DSC	0.9250	0.937*0*	A.6.5.4
NUHOMS®-32PTH1 DSC	0.9250	0.937*0*	A.6.5.4
NUHOMS®-37PTH DSC	0.9250	0.937*0*	A.6.5.7
NUHOMS®-61BT DSC	0.9364	0.9414	A.6.5.1
NUHOMS®-61BTH DSC NUHOMS®-61BTHF DSC	0.9400	0.9415	A.6.5.1
NUHOMS®-69BTH DSC	0.9389	0.9416	A.6.5.2

From a criticality analysis point of view, 32PTH and 32PTH Type 1 DSCs are identical. In this chapter, any reference to 32PTH is also applicable to 32PTH Type 1.

Due to the absence of any fissile material payload content in the NUHOMS®-irradiated/contaminated waste canister, no criticality calculations are required for this DSC. Therefore, no further discussion of the criticality of this canister is necessary.

A.6.2 Contents, Calculational Models and Criticality Calculations

The methodology employed to ensure the subcriticality of the 61BT, 61BTH, 24PT4 and 69BTH DSCs is based on a "fresh fuel" representation of the spent fuel assemblies. For these DSCs, the fuel assemblies are modeled with fresh (unirradiated) fuel.

The methodology employed to ensure the subcriticality of the 32PT, 24PTH, 32PTH, 32PTH1 and 37PTH DSCs is based on "burned fuel" representation of the spent fuel assemblies. Credit for the negative reactivity of the fuel assemblies as a result of irradiation, or "burnup credit" is employed in these calculations. *The maximum burnup "credited" in these analyses does not exceed 50 GWD/MTU.*

A.6.2.1 Fresh Fuel Methodology

For the NUHOMS®-61BT, 61BTH, 24PT4 and 69BTH DSCs, the system's criticality safety is ensured by both fixed neutron absorbers and favorable geometry. For each of these four DSCs, fresh fuel is assumed (no burnup credit is taken) in the evaluation. The fixed neutron absorber is present in the form of borated aluminum alloy or a boron-carbide/aluminum metal matrix

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28. *B. ROQUE, A. Santamarina, "Experimental Validation of Actinide and Fission Products Inventory from Chemical Assays in French PWR Spent Fuels," Practices and Developments in Spent Fuel Burnup Credit Applications, Proceedings of Technical Committee Meeting, Madrid, 22-26 April 2002, IAEA-TECDOC-1378, published October 2003.*

Table A.6-3
Modified Axial Burnup Profiles Used for SAS2H Depletion Analysis

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Table A.6-13
CSAS25 Benchmark Results

(continued)

Run ID	U Enrich. wt. %	Pu Enrich. Wt. %	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	EALF (eV)	k_{eff}	1σ
W3269SL1	2.72		1.524	1.494		0.3247	0.9973	0.0010
W3269SL2	5.7		1.422	1.93		0.3152	1.0024	0.0010
W3269W1	2.72		1.524	1.494		0.3080	0.9972	0.0012
W3269W2	5.7		1.422	1.93		0.3056	1.0015	0.0010
W3385SL1	5.74		1.422	1.932		0.2970	1.0004	0.0009
W3385SL2	5.74		2.012	5.067		0.1031	1.0014	0.0010
BAW1484A	2.46		1.636	1.84	1.636	0.1874	0.9942	0.0008
E196U6N	2.35		1.562	1.2		0.2578	0.9959	0.0008
E196U87C	2.35		2.21	3.69		0.0823	1.0011	0.0009
P2438X24	2.35		2.032	2.92	8.67	0.0944	0.9969	0.0008
SAXU56	5.74		1.4224	1.93		0.2909	0.9966	0.0011
SAXU792	5.74		2.0112	5.07		0.1023	0.9985	0.0010
EPRI70UN	0.71	2	1.778	1.2		0.7611	0.9983	0.0010
EPRI70B	0.71	2	1.778	1.2		0.5676	0.9999	0.0010
EPRI87B	0.71	2	2.210	1.53		0.2771	1.0077	0.0009
EPRI99UN	0.71	2	2.515	3.64		0.1355	1.0066	0.0009
EPRI99B	0.71	2	2.515	3.64		0.1798	1.0099	0.0009
SAXTON52	0.71	6.6	1.321	1.68		0.8858	1.0011	0.0010
SAXTON56	0.71	6.6	1.422	2.16		0.5404	1.0004	0.0012
SAXTN56B	0.71	6.6	1.422	2.16		0.6397	0.9997	0.0009
SAXTN735	0.71	6.6	1.867	4.7		0.1858	1.0019	0.0011
SAXTN792	0.71	6.6	2.012	5.67		0.1552	1.0026	0.0010
SAXTN104	0.71	6.6	2.642	10.75		0.1002	1.0051	0.0009
Correlation	**0.31**	**-0.37**	**0.46**	**0.25**	**0.63**	**0.01**	**N/A**	**N/A**
The Results of the CRC Benchmarks are shown below								
Run ID	**U Enrich. wt. %**	**Pu Enrich. Wt. %**	**Pitch (cm)**	**H_2O/fuel volume**	**Average Burnup (GWD/MTU)**	**EALF (eV)**	**k_{eff}**	**1σ**
CR3SP1	2.445		1.443	1.65	0	*0.5571*	*0.99377*	*0.00036*
CR3SP2	2.447		1.443	1.65	8.09	*0.6307*	*0.99166*	*0.00035*
CR3SP3	2.447		1.443	1.65	12.34	*0.6113*	*0.99413*	*0.00041*
CR3SP4	2.67		1.443	1.651	8.67	*0.6455*	*0.99063*	*0.00041*
CR3SP5	2.693		1.443	1.654	7.5	*0.6524*	*0.99346*	*0.00044*
CR3SP6	2.693		1.443	1.654	12.54	*0.6454*	*0.99289*	*0.00039*
CR3SP7	2.693		1.443	1.654	14.98	*0.6421*	*0.98935*	*0.0004*
CR3SP8	2.648		1.443	1.661	6.92	*0.6604*	*0.99149*	*0.0004*
CR3SP9	2.648		1.443	1.661	14	*0.6755*	*0.98797*	*0.00043*
CR3SP10	2.648		1.443	1.661	14.77	*0.6638*	*0.99547*	*0.00037*
CR3SP11	2.915		1.443	1.662	7.08	*0.7233*	*0.99355*	*0.0004*
CR3SP12	2.915		1.443	1.662	19.12	*0.7114*	*0.9952*	*0.00039*
CR3SP13	3.21		1.443	1.662	12.01	*0.7901*	*0.99355*	*0.00046*
CR3SP14	3.21		1.443	1.662	14.99	*0.7911*	*0.99454*	*0.0003*
CR3SP15	3.21		1.443	1.662	24.41	*0.7351*	*0.98907*	*0.00033*

Table A.6-13
CSAS25 Benchmark Results

(*continued*)

Run ID	U Enrich. wt. %	Pu Enrich. Wt. %	Pitch (cm)	H_2O/fuel volume	Average Burnup (GWD/MTU)	EALF (eV)	k_{eff}	1σ
CR3SP16	3.554		1.443	1.662	10.02	*0.8763*	*0.99127*	*0.00042*
CR3SP17	3.554		1.443	1.662	18.09	*0.8443*	*0.99072*	*0.0003*
CR3SP18	3.554		1.443	1.662	19.04	*0.8497*	*0.98937*	*0.0003*
CR3SP19	3.554		1.443	1.662	19.91	*0.8271*	*0.98818*	*0.00043*
CR3SP20	3.554		1.443	1.662	24.35	*0.7789*	*0.99095*	*0.00044*
CR3SP21	3.554		1.443	1.662	24.87	*0.7819*	*0.99104*	*0.00037*
CR3SP22	3.755		1.443	1.662	12.26	*0.9341*	*0.99005*	*0.00036*
CR3SP23	3.755		1.443	1.662	15.27	*0.9461*	*0.99057*	*0.00042*
CR3SP24	3.755		1.443	1.662	16.58	*0.9290*	*0.99007*	*0.00043*
CR3SP25	3.755		1.443	1.662	24.74	*0.8512*	*0.99016*	*0.00039*
CR3SP26	3.755		1.443	1.662	24.91	*0.8462*	*0.99118*	*0.00044*
CR3SP27	3.755		1.443	1.662	28.19	*0.8236*	*0.9871*	*0.00039*
CR3SP28	3.892		1.443	1.658	14.18	*0.9560*	*0.98832*	*0.00045*
CR3SP29	3.892		1.443	1.658	19.1	*0.9313*	*0.99224*	*0.00038*
CR3SP30	3.892		1.443	1.658	20.96	*0.9233*	*0.98937*	*0.0004*
CR3SP31	3.892		1.443	1.658	25.42	*0.8879*	*0.9862*	*0.00044*
CR3SP32	4.015		1.443	1.653	15.24	*1.0410*	*0.98086*	*0.00043*
CR3SP33	4.015		1.443	1.653	33	*0.8586*	*0.97859*	*0.00039*
NA1C5B	3.43		1.26	1.668	11.07	*0.9151*	*1.00482*	*0.00038*
SQ2C3BZ	3.43		1.26	1.668	*11.15*	*0.9606*	*1.00607*	*0.00035*
SQ2C3BF	3.43		1.26	1.668	*11*	*0.8651*	*1.00593*	*0.00039*
SQ2C3M	2.63		1.43	1.675	*19.25*	*0.9445*	*1.00569*	*0.00036*
SU1C2B	2.63		1.43	1.675	*6.93*	*0.6174*	*1.00538*	*0.00039*
SU1C2E	2.82		1.443	1.655	13.85	*0.6703*	*1.01023*	*0.00042*
TMI1C5B	3.54		1.26	1.668	*11.44*	*0.7276*	*1.00151*	*0.00037*
Correlation	**0.*168***		**0.*467***	**0.*256***	**-0.466**	**-0.*397***		

Notes:
The Cristal River benchmarks are labeled with a "CR3"and are based on 85 isotopes using the 238 Group Cross Section Library.

The Sequoyah benchmarks are labeled with a "SQ2", the Surry benchmarks are labeled with "SU1", the Three Mile Island benchmarks are labeled with "TMI1", the North Anna benchmarks are labeled with "NA1" and are based on 48 isotopes

Table A.6-13
CSAS25 Benchmark Results

(concluded)

Run ID	238 Group Library			44 Group Library			(238 to 44)
Run ID	K_{keno}	1σ	k_{eff}	K_{keno}	1σ	k_{eff}	Δk_{eff}
CR3SP1	0.99377	0.00036	0.9945	0.99649	0.00036	0.9972	-0.0034
CR3SP2	0.99166	0.00035	0.9924	0.995	0.00044	0.9959	-0.0042
CR3SP3	0.99413	0.00041	0.9950	0.99847	0.0004	0.9993	-0.0051
CR3SP4	0.99063	0.00041	0.9915	0.99449	0.00035	0.9952	-0.0046
CR3SP5	0.99346	0.00044	0.9943	0.99685	0.00036	0.9976	-0.0041
CR3SP6	0.99289	0.00039	0.9937	0.99689	0.00035	0.9976	-0.0047
CR3SP7	0.98935	0.0004	0.9902	0.99356	0.00039	0.9943	-0.0050
CR3SP8	0.99149	0.0004	0.9923	0.99494	0.00037	0.9957	-0.0042
CR3SP9	0.98797	0.00043	0.9888	0.99254	0.00041	0.9934	-0.0054
CR3SP10	0.99547	0.00037	0.9962	0.9995	0.00037	1.0002	-0.0048
CR3SP11	0.99355	0.0004	0.9944	0.99698	0.00039	0.9978	-0.0042
CR3SP12	0.9952	0.00039	0.9960	0.99977	0.00033	1.0004	-0.0052
CR3SP13	0.99355	0.00046	0.9945	0.99739	0.00035	0.9981	-0.0045
CR3SP14	0.99454	0.0003	0.9951	0.99868	0.0003	0.9993	-0.0047
CR3SP15	0.98907	0.00033	0.9897	0.99412	0.00032	0.9948	-0.0057
CR3SP16	0.99127	0.00042	0.9921	0.99533	0.00036	0.9961	-0.0048
CR3SP17	0.99072	0.0003	0.9913	0.99466	0.0003	0.9953	-0.0045
CR3SP18	0.98937	0.0003	0.9900	0.9938	0.00029	0.9944	-0.0050
CR3SP19	0.98818	0.00043	0.9890	0.99318	0.00036	0.9939	-0.0057
CR3SP20	0.99095	0.00044	0.9918	0.99514	0.00036	0.9959	-0.0049
CR3SP21	0.99104	0.00037	0.9918	0.99512	0.00037	0.9959	-0.0048
CR3SP22	0.99005	0.00036	0.9908	0.99431	0.00037	0.9951	-0.0050
CR3SP23	0.99057	0.00042	0.9914	0.99366	0.00043	0.9945	-0.0040
CR3SP24	0.99007	0.00043	0.9909	0.99506	0.00041	0.9959	-0.0058
CR3SP25	0.99016	0.00039	0.9909	0.99486	0.00038	0.9956	-0.0055
CR3SP26	0.99118	0.00044	0.9921	0.99576	0.00039	0.9965	-0.0054
CR3SP27	0.9871	0.00039	0.9879	0.99117	0.0004	0.9920	-0.0049
CR3SP28	0.98832	0.00045	0.9892	0.99321	0.00035	0.9939	-0.0056
CR3SP29	0.99224	0.00038	0.9930	0.99611	0.00034	0.9968	-0.0046
CR3SP30	0.98937	0.0004	0.9902	0.99298	0.00041	0.9938	-0.0044
CR3SP31	0.9862	0.00044	0.9871	0.98904	0.00035	0.9897	-0.0035
CR3SP32	0.98086	0.00043	0.9817	0.98495	0.00039	0.9857	-0.0049
CR3SP33	0.97859	0.00039	0.9794	0.98309	0.00037	0.9838	-0.0052
NA1C5B	1.00482	0.00038	1.0056	1.00897	0.00041	1.0098	-0.0050
SQ2C3BZ	1.00607	0.00035	1.0068	1.01229	0.00036	1.0130	-0.0069
SQ2C3BF	1.00593	0.00039	1.0067	1.01028	0.00037	1.0110	-0.0051
SQ2C3M	1.00569	0.00036	1.0064	1.00995	0.00041	1.0108	-0.0051
SU1C2B	1.00538	0.00039	1.0062	1.00776	0.00034	1.0084	-0.0031
SU1C2E	1.01023	0.00042	1.0111	1.01529	0.00038	1.0161	-0.0058
TMI1C5B	1.00151	0.00037	1.0023	1.00507	0.00038	1.0058	-0.0043

Table A.6-14
USLSTATS Results

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Table A.6-15
USL Determination for Criticality Analysis

Proprietary Information Withheld Pursuant to 10 CFR 2.390

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2. Verify that the cask surface removable contamination levels meet the requirements of 49 CFR 173.443 [2] and 10 CFR 71.87 [3].
3. Verify that the assembly verification leakage rate testing specified in Section A.7.4.1 has been performed. This test must be performed within 12 months prior to the shipment. This test includes drawing a vacuum on the cavity between the cask and DSC, which ensures that any water has been removed.

A.7.1.5.1 Placing the Cask onto the Conveyance

The procedure for placement of the cask on the conveyance is given in this section. If cask is already on the transportation skid, rig the cask/skid, lift and place them on to the conveyance, then skip to Step 8.

1. Using a suitable prime mover, bring the cask and onsite transfer trailer to the conveyance.
2. Remove the onsite transfer skid pillow block covers.
3. Install suitable slings around the cask top and bottom ends.
4. Lift the cask from the onsite transfer trailer.
5. Remove the cask upper and lower trunnions and install the trunnion plugs.
6. Place the cask onto the transportation skid.
7. Remove the lifting slings from the cask.
8. If necessary, install the external aluminum fins.
9. Install the transportation skid closure assembly, if not already installed.
10. Install the impact limiters on the cask and torque the attachment bolts in accordance with the drawings in Chapter A.1, Appendix A.1.4.10.1.
11. Remove the impact limiter hoist rings and replace them with hex bolts provided prior to transport.
12. Install the cask tamperproof seals.
13. *Install the transportation skid personnel barrier.*
14. Monitor the cask radiation levels per 49 CFR 173.441 [2] and 10 CFR 71.47 [3] requirements.
15. Verify that the temperature on all accessible surfaces is < 185°F.
16. Prepare the final shipping documentation and release the loaded cask for shipment.

Enclosure 7 to TN E-28237

B. ROQUE, A. Santamarina, "Experimental Validation of Actinide and Fission Products Inventory from Chemical Assays in French PWR Spent Fuels," Practices and Developments in Spent Fuel Burnup Credit Applications, Proceedings of Technical Committee Meeting, Madrid, 22-26 April 2002, IAEA-TECDOC-1378, published October 2003.

IAEA-TECDOC-1378

Practices and developments in spent fuel burnup credit applications

Proceedings of an Technical Committee meeting held in Madrid, 22–26 April 2002

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

October 2003

Experimental validation of actinide and fission products inventory from chemical assays in French PWR spent fuels

B. Roque, A. Santamarina
CEA - Cadarache, DRN/DER/SPRC, Saint Paul lez Durance, France

Abstract. In order to validate fuel inventory calculations, a large experimental programme has been performed in France, based on spent fuel chemical assays. Experimental data are based on chemical analysis measurements from fuel rod cuts irradiated in French PWR reactors and from full assembly dissolutions at the COGEMA/La Hague reprocessing plants. This enables us to cover a large range of UOx fuels with various enrichments in ^{235}U, 3.1% to 4.5%, associated with burnups from 10 GWd/t to 60 GWd/t. Uranium, Plutonium, Americium and Curium isotopes were analysed in PWR samples. Furthermore, Fission Products involved in Burn up Credit studies were measured. Calculation to Experiment comparison has been obtained using the APOLLO2/DARWIN package developed by the CEA. The results are described and discussed in the paper.

1. Introduction

In order to validate fuel inventory calculations, a large experimental programme based on spent fuel chemical analysis has been carried out in France since 1993. Uranium, plutonium, neptunium, americium and curium isotopes have been analysed in PWR samples. Furthermore the 15 fission products selected by the OECD for Burn up Credit (BUC) criticality calculations have been measured.

The available experimental information consists of chemical assays from fuel rod cuts irradiated in French PWR reactors and of solution samples derived from full assembly dissolutions at the COGEMA/La Hague reprocessing plants.

The calculation to experiment comparison have been obtained with the DARWIN package, developed by the CEA and its French industrial partners, and devoted to fuel cycle studies.

2. Presentation of French Tools

The fuel inventory validation is carried out using the DARWIN package described here after, based for PWR studies on the APOLLO2 code and the depletion code PEPIN2.

The DARWIN package

DARWIN is the reference calculation package for the fuel cycle of all types of reactors. It was developed by the CEA and its French partners (COGEMA, EDF and FRAMATOME) to estimate the physics quantities characterizing the burnup fuels from reactors: material balance, decay heat, activity, neutron, γ, α, β sources and spectrum, radiotoxicity.

DARWIN is devoted to all cycle studies, with current fuels (UOx, MOx) or innovative fuels (MIX, APA, PuTh) and for every nuclear road (Pressured Water Reactor, Fast Breeder Reactor, Boiling Water Reactor, Advanced Reactors). DARWIN is also used in the back-end cycle for actinide incineration (SPIN) or long term interim storage studies.

The simplified DARWIN structure, based on new codes and libraries is described in the Figure 1.

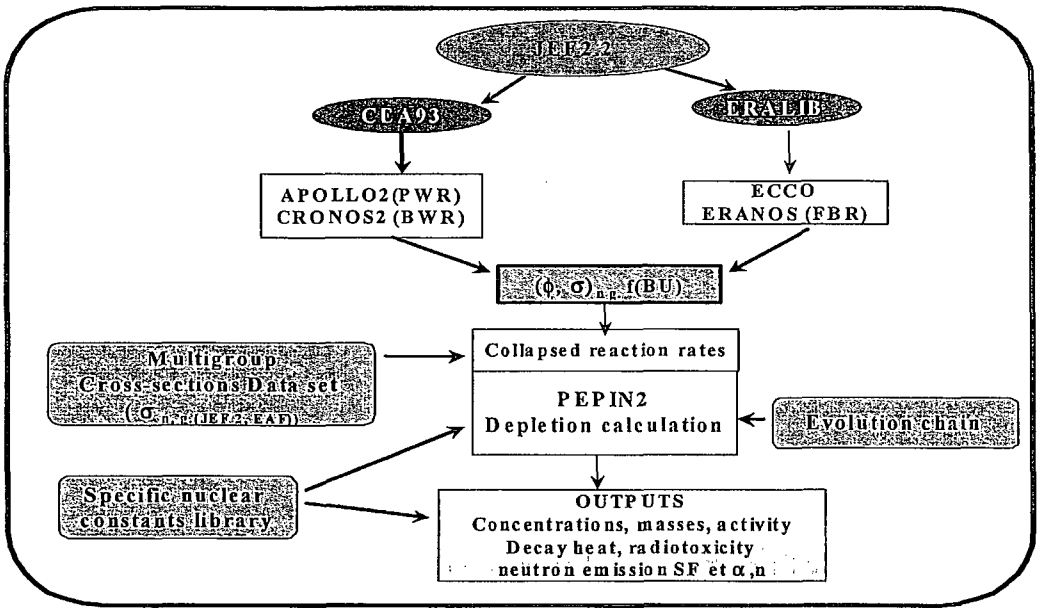

FIG. 1. The DARWIN package.

The PEPIN2 program performs the nuclide depletion calculations. Different libraries feed this module

— neutronics data provided by French assembly transport codes APOLLO2 (for PWR studies), ECCO-ERANOS system (for FBR studies) : these data are self-shielded cross sections and neutron spectra;
— nuclear data (decay data, fission and □, n) yields) and evolution chains;
— complementary cross-sections, missing from the transport codes libraries, specially for activation products. They are included in the 'cycle library'.

The basic nuclear data comes from the JEF2.2 European evaluation; in the scope of our study on PWR assemblies, the neutronic data necessary to the depletion module are provided by the APOLLO2 code and its CEA93 library and benefits from its extensive experimental validation achieved in the framework of the PWR lattices.

DARWIN makes possible the retrieval of cumulated reaction rates during irradiation in order to give the origin of every isotope build-up. Furthermore, a "PERTURBATION" of main nuclear data, such as capture cross-section, initial isotopic composition, flux, is also available and allows sensitivity studies.

3. Calculational models

The accuracy of the DARWIN results depends mainly on the APOLLO2 assembly calculation. APOLLO2 solves the integral form of the Boltzmann equation through the collision probability method. We use an APOLLO2 reference calculation route devoted to depletion calculation of 17X17 UO2 PWR assemblies. This calculation route uses the CEA93 cross-section library in 172 groups structure processed from the JEF2 evaluations. Among the themes studied in the optimisation of the assembly calculation, we can mention:

- the radial discretization of the fuel pellet in 4 rings in order to give a faithful representation of the resonant absorption of U238 in the pin and of the actinide and fission product concentration profile;
- the spatial calculation achieved using the UP1 HETE approximation. This enables the probability of leakage PIS and the probability of transmission PSS to be calculated for the true geometry, the interface currents are considered to be linearly anisotropic;
- a grouping of cells with similar flux within a unique 'physical cell';
- the self-shielding of resonant isotopes, with a differentiated treatment for each one depending on the physical characteristics of their cross-sections;
- optimized evolution steps.

The PEPIN2 evolution module then uses the results provided by APOLLO2, self-shielded cross-sections and multigroup spectra - to make up the collapsed library with burnup dependent cross-sections, required in order to characterize the isotopes described in the depletion chains.

4. The experimental database of irradiated PWR fuels

We describe here the main experimental programmes carried out on irradiated UO_X PWR fuels.

The experimental information can be divided into two categories:

- small samples of fuel pins, irradiated in French reactors, with positions in the assembly well characterized. These are sensitive to local irradiation conditions. These time-consuming and expensive experiments provide very accurate results for a limited number of samples;
- dissolution aliquots of entire assembly sets; these are numerous and very different as to the type of assemblies covered, but the information on the irradiation condition is limited.

Four programmes related to fuel samples are used for the experimental validation of actinides and fission products inventory: BUGEY3, FESSENHEIM II, GRAVELINES and CRUAS.

- **BUGEY3** (900 MWe) uses standard fuel with 3.1% initial enrichment and the assembly consists of 17X17 rods with a zircaloy clad. The maximum burnup is 40 GWd/t.
- **FESSENHEIM II** allows the study of UOx fuel (3.1% 235U) with high burnup up to 60 GWd/t.
- **GRAVELINES** is devoted to the extension of the calculation scheme validation for high burnup (five irradiation cycles) with higher enrichment corresponding to 4.5%. This experimental programme on a 900 MWe PWR is the most important being carried out in France today.
- **CRUAS** is devoted to the validation of the URE (Uranium Reprocessed Enriched) fuel, using reprocessed then enriched uranium. It allows the ^{236}U capture validation.

From full assembly dissolutions at COGEMA/La Hague reprocessing plants, we get uranium and plutonium chemical analyses. The assembly involved are 17X17 PWR at 3.1%, 3.25% and 3.45% enrichment with burnup between 25 and 40 GWd/t.

5. Results of the experimental validation

The calculation-experiment comparison (C-E)/E (in %) is summarised below for actinides and important fission products, including the BUC fission products.

The calculation burnup is adjusted using an experimental indicator, namely the $^{145}Nd/^{238}U+^{146}Nd/^{238}U$ sum for PWR fuel cut analyses and the $^{235}U/^{238}U$ residual enrichment for the assembly dissolutions (the neodymium chemical analyses are not available).

In the following Table the total uncertainties correspond to the combination of uncertainties on chemical assays and determination of the burnup of the assembly derived from Nd (or ^{235}U) isotopics. When more than one sample is available for the same burnup, the spread of results is also considered.

It is important to remind that for burnup credit studies an underestimation of absorbing fission products buildup or an overestimation of fissile isotopes are conservative, but respectively an overestimation and an underestimation are not.

5.1. Uranium isotope results

Table I points out that uranium isotope concentration are well predicted for Burn up Credit application.

- The ^{235}U depletion is accurately simulated up to very high burnup, even if the quantities involved become very small, and the associated uncertainty and the sensitivity to the cross-section become greater. For the "La Hague" results, the (C-E)/E is negligible because the calculation burnup is adjusted using the $^{235}U/^{238}U$ residual enrichment.
- The abundance of ^{236}U is underestimated by -4% in standard UOx fuel, principally due to the underestimation of the ^{235}U capture Resonance Integral in JEF2. Many works have been done on this subject and a new evaluation, called Leal-Derrien-Wright-Larson will be introduced in the European JEFF3.0 file. The ^{236}U build-up will be very well predicted: (C-E)/E = -1% ± 1.0% at 60 GWd/t. In URE fuel, the underestimation is no more existing because the ^{236}U concentration is mainly linked, at low burnup, on the initial ^{236}U content and not on the ^{235}U capture.
- The abundance of ^{234}U is well reproduced by calculation if the initial content in the fresh fuel is well known. This is the case, for example for the CRUAS experiment.

Table I. (C-E)/E (%) for 'Uranium' Inventory

Fuel↓	*BU (GWd/t)* ↓	$^{234}U/^{238}U$	$^{235}U/^{238}U$	$^{236}U/^{238}U$
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	2.5 ± 0.6	0.4 ± 1.3	-3.6 ± 1.0
	25	5.2 ± 1.1	0.3 ± 2.4	-3.1 ± 1.2
	40	1.7 ± 0.7	0.04 ± 2.1	-3.2 ± 0.4
	50	-5.9 ± 1.1	0.1 ± 2.7	-4.2 ± 0.1
	60	-2.3 ± 2.3	3.3 ± 7.5	-4.1 ± 0.2
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-0.3 ± 2.0	-0.7 ± 1.7	-3.4 ± 1.3
	40	0.2 ± 2.2	-0.4 ± 2.8	-3.9 ± 1.0
	50	-4.7 ± 2.2	1.4 ± 4.1	-4.9 ± 0.6
	60	1.4 ± 2.5	0.8 ± 5.3	-4.3 ± 0.3
UOx "La HAGUE" 3.1 % ^{235}U	*30*	-16.6 ± 2.2	-0.01 ± 2.8	-1.4 ± 1.0
	40	-15.7 ± 2.2	0.03 ± 4.1	-1.2 ± 0.6
UOx "La HAGUE" 3.25 % ^{235}U	*35*	-12.5 ± 2.2	-0.01 ± 2.8	-2.4 ± 1.0
	45	-13.7 ± 2.2	-0.01 ± 2.8	-1.8 ± 1.0
UOx "La HAGUE" 3.45 % ^{235}U	*35*	-16.5 ± 2.2	-0.05 ± 2.8	-4.7 ± 1.0
URE "CRUAS" 3.5 % ^{235}U	*15*	0.2 ± 0.5	-0.2 ± 1.0	-0.5 ± 0.4
	25	0.2 ± 0.8	0.8 ± 1.7	-0.3 ± 0.3
	35	-0.5 ± 1.1	0.02 ± 3.1	-1.1 ± 0.2

5.2. *Plutonium isotope results*

Table II. (C-E)/E (%) for 'Plutonium' Inventory

Fuel↓	*BU(GWd/t)* ↓	$^{238}Pu/^{238}U$	$^{239}Pu/^{238}U$	$^{240}Pu/^{238}U$	$^{241}Pu/^{238}U$	$^{242}Pu/^{238}U$
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-12.3 ± 3.2	-0.6 ± 1.1	-0.9 ± 1.6	-4.0 ± 2.1	-6.6 ± 3.9
	25	-9.0 ± 4.6	-0.1 ± 1.5	-0.7 ± 2.1	-3.2 ± 2.9	-5.4 ± 5.0
	40	-6.4 ± 2.0	2.5 ± 0.9	-1.9 ± 0.8	-1.0 ± 1.2	-7.6 ± 2.2
	50	-4.2 ± 1.6	2.0 ± 1.1	-1.2 ± 0.6	-0.4 ± 1.2	-6.1 ± 1.8
	60	-8.7 ± 2.5	3.1 ± 2.4	-0.7 ± 0.9	-0.3 ± 2.6	-8.7 ± 2.9
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-12.2 ± 5.3	-1.6 ± 0.6	-2.2 ± 2.3	-5.0 ± 2.7	-6.8 ± 5.5
	40	-10.0 ± 5.0	-0.4 ± 0.2	-1.9 ± 2.0	-4.2 ± 2.0	-6.9 ± 4.8
	50	-10.3 ± 4.5	+1.6 ± 0.1	-1.8 ± 1.5	-3.0 ± 1.2	-8.6 ± 4.3
	60	-9.9 ± 4.3	+1.9 ± 0.2	-1.5 ± 1.2	-2.6 ± 1.0	-7.2 ± 3.9
UOx "La HAGUE" 3.1 % ^{235}U	*30*	-14.9 ± 5.0	+0.9 ± 0.2	-1.7 ± 2.0	-2.5 ± 2.0	-11.7 ± 4.8
	40	-15.4 ± 4.5	+0.7 ± 0.1	-1.0 ± 1.5	-6.7 ± 1.2	-11.9 ± 4.3
UOx "La HAGUE" 3.25 % ^{235}U	*35*	-14.1 ± 5.0	1.1 ±0.2	-1.7 ± 2.0	-1.9 ± 2.0	-11.1 ± 4.8
	45	-11.7± 4.5	2.1 ± 0.1	-0.5 ± 1.5	-0.5 ± 1.2	-9.5 ± 4.3
UOx "La HAGUE" 3.45 % ^{235}U	*35*	-18.4 ± 5.0	-0.1 ± 0.2	-2.5 ± 2.0	-2.6 ± 2.0	-11.5 ± 4.8
URE "CRUAS" 3.5 % ^{235}U	*15*	-2.8 ± 3.9	1.7 ± 1.2	-2.2 ± 2.6	0.5 ± 3.6	-5.1 ± 5.8
	25	-2.8 ± 3.4	1.0 ± 1.0	-0.6 ± 2.2	-1.6 ± 2.7	-4.9 ± 5.1
	35	-1.7 ± 3.3	1.9 ± 1.0	-0.5 ± 1.7	-0.6 ± 1.8	-4.2 ± 4.3

— The abundance of ^{238}Pu is underestimated by about 10% in standard UOx fuels. However, in BUC application, this underestimation is not very penalizing because the ^{238}Pu is not a great contributor to the actinide BUC negative reactivity worth (10% of the total minor actinide reactivity worth at 40 GWd/t).

- The ^{239}Pu is well predicted within 2% accuracy. The slight overestimation increasing with burnup could indicate an underestimation of its (n, γ) cross-section in JEF2. This overestimation is conservative for BUC studies.
- The prediction of ^{240}Pu is very well estimated, confirming the correct modelling of Doppler/self-shielding resonance for 240Pu at 1 eV. The negative reactivity worth of this most poisoning isotope will be very well represented in our BUC calculations.
- ^{241}Pu is slightly underestimated. Its prediction is highly sensitive to ^{239}Pu capture. This result could point out again a slight under-estimation of ^{239}Pu (n, γ) cross-section in JEF2 evaluation. We must taking into account this underestimation leading to a non-conservative BUC calculation.
- ^{242}Pu is underestimated. This result shows that ^{241}Pu capture cross-section in JEF2.2 is under-estimated. Hence, a new evaluation of Pu^{241} will be introduced in JEFF3.0: the increase of the capture will correct the ^{242}Pu, ^{243}Am and ^{244}Cm build-up.

5.3. Minor actinides results

Table III. (C-E)/E (%) for 'Neptunium' Inventory

Fuel↓	*BU (GWd/t)*↓	$^{237}Np/^{238}U$
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-10.5 ± 4.4
	25	-7.2 ± 4.3
	40	-0.6 ± 2.1
	50	-3.2 ± 2.5
	60	3.0 ± 3.1
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-0.3 ± 4.7
	40	-2.4 ± 4.2
	50	-4.4 ± 3.4
	60	-3.0 ± 3.1
URE "CRUAS" 3.5 % ^{235}U	*15*	-4.1 ± 3.2
	25	-0.2 ± 3.1
	35	3.9 ± 3.0

The abundance of ^{237}Np is underestimated, due on one hand to the underestimation of the ^{235}U capture Resonance Integral and in the other hand to the underestimation of $(n,2n)^{238}U$ in JEF2.

However, the accuracy of our calculations is conservative for BUC UOx applications.

- The ^{241}Am, formed by decaying of ^{241}Pu, is reasonably estimated for BUC UOx applications.
- ^{242}Am is underestimated by about -20%. This suggests increasing ^{241}Am capture cross-section. We must also improve the knowledge of the branching ratio of ^{241}Am to ^{242}Am in the epithermal range.
- The underestimation in ^{242}Pu generates the underestimation for ^{243}Am.

Table IV. (C-E)/E (%) for 'Americium' Inventory

Fuel↓	*BU (GWd/t)*↓	$^{241}Am/^{238}U$	$^{242m}Am/^{238}U$	$^{243}Am/^{238}U$
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-6.3 ± 3.4	-25.2 ± 10.1	-10.6 ±7.8
	25	-2.9 ± 3.1	-9.9 ± 3.7	-10.6± 7.5
	40	-3.0 ± 1.2	-17.3 ± 3.0	-4.5 ± 3.4
	50	-13.2 ± 2.6		-10.5 ± 2.9
	60	-3.9 ± 2.4	-21.1 ± 11.4	-8.8 ± 4.6
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-5.0 ± 2.8	-33.0 ± 5.0	-10.3 ± 7.7
	40	-3.5 ± 2.1	-25.0 ± 3.2	-7.9 ± 7.0
	60	+1.3 ± 1.0	-16.2 ± 2.4	-4.8 ± 5.6
URE "CRUAS" 3.5 % ^{235}U	*35*	-1.3 ± 2.3	-17.0 ± 2.8	-4.8 ± 6.4

Table V. (C-E)/E (%) for 'Curium' Inventory

Fuel↓	*BU (GWd/t)*↓	$^{243}Cm/^{238}U$	$^{244}Cm/^{238}U$	$^{245}Cm/^{238}U$	$^{246}Cm/^{238}U$	$^{247}Cm/^{238}U$
UOX "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-42.7 ± 11.3	-33.7 ± 8.0	-27.5 ± 10.4	-81.2 ± 8.7	/
	25	-34.5 ± 8.9	-19.9 ± 9.1	-19.2 ± 10.9	-7.7 ± 17.8	/
	40	-9.6 ± 5.8	-12.1 ± 5.4	-2.3 ± 7.8	-27.8 ± 7.2	/
	50	-28.3 ± 10.2	-21.6 ± 8.1	-18.2 ± 13.4	-31.0 ± 11.3	/
UOX "GRAVELINES" 4.5 % ^{235}U	*30*	-28.1 ± 8.0	-18.5 ± 10.4	-20.0 ± 12.4	-25.0 ± 15.5	-25.6 ± 18.2
	40	-26.5 ± 8.0	-15.4 ± 9.8	-16.8 ± 11.5	-23.0 ± 15.2	-30.2 ± 18.0
	60	-24.5 ± 8.0	-12.4 ± 8.4	-11.4 ± 9.3	-23.1 ± 13.4	-30.2 ± 16.3
URE "CRUAS" 3.5 % ^{235}U	*35*	-5.4 ± 6.3	-2.03 ± 9.3	-0.05 ± 11.2	-12.2 ± 12.8	-12.9 ± 31.7

The curium isotopes are underestimated. The underestimation in ^{242}Pu generates an underestimation for ^{243}Am, and consequently for ^{244}Cm and ^{245}Cm. This underestimation is acceptable for BUC studies on UOx fuels but recent studies on ^{241}Pu JEF2 capture cross-section enable to partially correct this problem.

The underestimation in ^{243}Cm build-up needs also to be corrected. Studies on ^{242}Cm capture cross-section are in progress.

The concentrations of the most important major and minor actinides, poisoning such as ^{236}U, ^{237}Np, ^{240}Pu, ^{241}Am, or fissile as ^{235}U, ^{239}Pu, ^{241}Pu are well calculated and can be used directly in the CRISTAL package for BUC calculations in UOx fuels; the slight underestimation of ^{241}Pu build-up, leading to a non-conservative calculation, needs to be taking into account by the way of a correction factor on its concentration.

5.4. Fission products results

5.4.1. The samarium isotopes

Table VI. (C-E)/E (%) for 'Samarium' Inventory

Fuel↓	*BU (GWd/t)↓*	147Sm/238U	149Sm/238U	150Sm/238U	151Sm/238U	152Sm/238U
UOX "BUGEY" 3.1 % ^{235}U	*20*	-4.9 ± 1.5	4.5 ± 4.7	-3.7 ± 2.6	-3.1 ± 1.2	0.4 ± 1.9
	40	-5.9 ± 1.0	-3.4 ± 23.3	-3.6 ± 2.2	7.3 ± 1.5	1.1 ± 1.7
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	-6.5 ± 1.1	-8.6 ± 8.4	-3.2 ± 2.3	3.6 ± 1.1	1.4 ± 1.8
	50	-7.0 ± 1.0	5.6 ± 4.8	-4.8 ± 2.1	7.9 ± 1.1	3.5 ± 1.6
	60	-7.4 ± 0.9	-3.5 ± 10.0	-4.1 ± 1.8	12.8 ± 1.0	5.6± 1.7

Except the slight overestimation of ^{151}Sm build-up, the samarium isotope calculations are satisfactory.

- The ^{147}Sm is underestimated. This isotope is formed by decaying of ^{147}Pm, itself formed by cumulative fissions, and disappeared by neutron capture. The interpretation of ^{147}Sm samples oscillations in the MINERVE reactor shows that the ^{147}Sm capture cross-section is well calculated (C-E)/E = -0.3% ± 3.2%; furthermore the sensitivity of ^{147}Sm build-up is very low because of the low value of the cross-section (57 barns at 2200m/s); so we can conclude that the underestimation of ^{147}Sm is due to its formation. The capture cross-section of ^{147}Pm and ^{147}Nd cumulative fission yields must be investigated.
- The ^{149}Sm seems to be underestimated but the chemical uncertainties is very high and the ^{149}Sm build-up is very sensitive to the power history mesh. However this slight underestimation is confirmed by rigorous APOLLO2 calculation describing precisely the irradiation history.
- The ^{150}Sm is underestimated, varying in the direct ratio of its father the ^{149}Sm.
- ^{151}Sm is overestimated. This is probably due to an overestimation of ^{151}Pm cumulative fission yields.

— ^{152}Sm is well predicted but we can notice a slow drift toward overestimation when the burnup increases. This could be due to ^{151}Sm calculation overestimation but also to an underestimation of the ^{152}Sm capture cross-section as suggested by the MINERVE oscillation experiment. More studies are needed to conclude.

5.4.2. The neodymium isotopes

Table VII. (C-E)/E (%) for 'Neodymium' Inventory

Fuel↓	*BU (GWd/t)*↓	$^{143}Nd/^{238}U$	$^{144}Nd/^{238}U$	$^{148}Nd/^{238}U$	$^{150}Nd/^{238}U$
UOX "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	0.3 ± 1.3	-2.0 ± 1.8	0.2 ± 1.7	-6.7 ± 1.8
	25	0.3 ± 1.7	-2.2 ± 2.5	1.5 ± 2.2	0.7 ± 2.5
	40	/	/	0.2 ± 1.7	/
	50	2.4 ± 1.8	0.2 ± 2.9	-0.6 ± 2.6	-0.8 ± 3.2
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	0.5 ± 1.4	-1.9 ± 2.5	1.6 ± 2.1	0.7 ± 2.4
	50	1.6 ± 1.0	-2.6 ± 2.6	1.5 ± 2.1	0.5 ± 2.4
	60	1.9 ± 0.4	-2.4 ± 1.5	1.4 ± 1.2	0.4 ± 1.4
URE "CRUAS" 3.5 %	*15*	0.0 ± 1.8	-1.8 ± 2.1	0.7 ± 2.1	-0.3 ± 2.3
	25	0.6 ± 1.6	-2.1 ± 2.2	0.7 ± 2.1	-0.1 ± 2.2
	35	0.7 ± 1.2	-2.3 ± 2.3	1.3 ± 2.1	0.6 ± 2.3

The Nd isotopes are very well calculated.

In Table VII, the ^{143}Nd is the unique FP involved in BUC studies. Its build-up is slightly overestimated for high burnups : +2%. At these burnups, ^{143}Nd concentration is sensitive to its capture cross-section. This result confirms an under-estimation of the thermal capture cross-section of ^{143}Nd. The same conclusion is drawn from the interpretation of ^{143}Nd samples oscillated in Minerve lattices. A new evaluation of ^{143}Nd will be introduced in the future JEFF3 file, in order to increase by 4% the (n, γ) cross-section in the 0-0.2eV energy range

One can notice a slight underestimation of ^{144}Nd concentration due probably to an underestimation of ^{144}Ce cumulative fission yields.

The most absorbing FP, involved in BUC studies, is the ^{133}Cs. A slight and steady under-prediction by -4% is noticed. This is due to an underestimation of ^{133}Xe fission yields for ^{235}U and ^{239}Pu in the JEF2.2 library.

^{137}Cs is not an absorbing FP but it is used as a burnup monitor and must be calculated with about 2% accuracy. Our calculation leads to an under-prediction of the ^{137}Cs concentration ; this is due to underestimation of ^{137}Cs fission yields in JEF2.2.

5.4.3. The cesium isotopes

Table VIII. (C-E)/E (%) for 'Cesium' Inventory

Fuel↓	*BU (GWd/t)*↓	$^{133}Cs/^{238}U$	$^{134}Cs/^{238}U$	$^{135}Cs/^{238}U$	$^{137}Cs/^{238}U$
UOX "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-2.7 ± 2.0	-8.1 ± 4.6	18.8 ± 0.6	-1.8 ± 2.3
	25	-3.9 ± 1.9	-14.8 ± 4.8	15.2 ± 0.9	-4.0 ± 2.2
	40	-8.5 ± 1.8	-9.3 ± 4.9	3.9 ± 0.7	-1.2 ± 2.3
	50	-1.3 ± 1.6	-3.7 ± 5.6	5.1 ± 0.9	-4.4 ± 2.2
	60	-2.2 ± 1.5	2.1 ± 6.1	1.8 ± 0.9	-3.6 ± 2.3
UOX "GRAVELINES" 4.5 % ^{235}U	*30*	-4.9 ± 2.0	-11.7 ± 4.0	-1.9 ± 1.9	-5.7 ± 2.2
	40	-3.3 ± 2.0	-10.4 ± 3.8	-2.4 ± 2.1	-5.0 ± 2.3
	50	-3.9 ± 1.8	-8.2 ± 3.5	-3.9 ± 2.2	-6.2 ± 2.3
	60	-3.4 ± 1.2	-8.7 ± 2.9	-3.3 ± 1.6	-5.2 ± 1.5

5.4.4. The metallic fission product isotopes

Table IX. (C-E)/E (%) for 'Metallic Fission Products' Inventory

Fuel ↓	*BU (GWd/t)* ↓	$^{99}Tc/^{238}U$	$^{95}Mo/^{238}U$	$^{101}Ru/^{238}U$	$^{103}Rh/^{238}U$
UOX "BUGEY" 3.1 % ^{235}U	*20*	-7.6 ± 3.1	-3.7 ± 2.7	2.9 ± 2.9	3.5 ± 2.8
	40	4.8 ± 3.0	8.4 ± 2.6	15.2 ± 2.9	13.9 ± 2.6
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	-0.6 ± 3.1	5.5 ± 2.7	5.0 ± 2.9	3.4 ± 2.7
	50	2.6 ± 3.1	2.6 ± 2.6	6.4 ± 2.3	4.6 ± 2.5
	60	3.6 ± 3.0	2.1 ± 2.6	0.9 ± 2.9	3.0 ± 2.3

Metallic fission products isotopic prediction is quite satisfactory in UOx spent fuels since they are in agreement with the experimental standard deviation. However, it should be noted that the dissolution of metallic fission product such as ^{95}Mo, ^{99}Tc, ^{101}Ru and ^{103}Rh could lead to non-soluble deposits. A new programme of dissolution, more accurate, is planned in order to confirm the C/E discrepancies.

5.4.5. The europium and gadolinium isotopes

In order to check ^{155}Gd build-up which is an important poisoning FP in BUC, the ^{154}Eu, ^{155}Eu and ^{154}Gd were also investigated. DARWIN results are presented in Table X.

Table X. (C-E)/E (%) for 'Europium And Gadolinium' Inventory

Fuel↓	*BU (GWd/t)* ↓	^{153}Eu/^{238}U	^{154}Eu/^{238}U	^{155}Eu/^{238}U	^{154}Gd/^{238}	^{155}Gd/^{238}U	^{156}Gd/^{238}U
UOX "BUGEY" 3.1 % ^{235}U	*20*	6.5 ± 3.0	40.9 ± 4.6	8.9 ± 3.8	40.3 ±4.5	-3.1 ± 3.4	
	40	11.8 ± 2.6	80.4 ± 4.2	8.9 ± 4.2	82.4 ±3.8	0.1 ± 3.7	
UOX "GRAVELINES " 4.5 % ^{235}U	*40*	8.9 ± 3.0	54.2 ± 6.4	13.9 ± 4.2	/	4.3 ± 3.8	
	50	11.7 ± 2.8	74.8 ± 6.6	15.4 ± 4.5	72.6 ± 6.0	8.1 ± 3.9	-22.0 ± 3.9
	60	16.4 ± 2.5	94.0 ± 6.2	18.2 ± 4.4	/	11.9 ± 4.0	-15.0 ± 4.0

^{155}Gd is strongly over-predicted but agrees with its father, ^{155}Eu, which is produce by capture on ^{154}Eu and directly by fission. Furthermore the ^{156}Gd, daughter of ^{155}Eu is under-predicted.

^{154}Eu is also strongly overestimated and the same level of overestimation is found on its daughter ^{154}Gd. These remarks point out that modification of ^{154}Eu and ^{155}Eu capture cross-section is needed in JEF2 file.

Recent studies on ^{154}Eu and ^{155}Eu capture cross-section confirm that data used in JEF2.2 file are not satisfactory and will be replaced by ENDFB6 evaluations in the new JEFF3.0 file. The use of these evaluations leads to the following results on Eu and Gd build-up in GRAVELINES UOx fuels.

Table XI. New (C-E/E) (%) results using σc from ENDFB6.7 for ^{154}EU and ^{155}EU in GRAVELINES UOx fuels

Burnup (GWd/t)	*40*	*50*	*60*
^{154}Eu/^{238}U	-7.1%	-4.1%	0.8%
^{155}Eu/^{238}U	13.6%	13.5%	15.5%
^{154}Gd/^{238}U		-3.9%	
^{155}Gd/^{238}U	4.0%	6.2%	9.1%
^{156}Gd/^{238}U		-3.8 %	2.4 %

The use of both europium 154 and europium 155 capture cross-section from the ENDFB6.7 evaluation leads to a better gadolinium 155 calculation build-up but a residual overestimation remains; this could be due to overestimation of ^{155}Sm cumulative fission yields (leading to the overestimation of ^{155}Eu) and/or to an underestimation of the capture cross-section of the ^{155}Gd itself.

The ^{154}Eu concentration, used as a burnup monitor for long cooling time, is well improved with a slight underestimation; the conclusion is the same for ^{154}Gd.

The ^{156}Gd calculation is also well improved by the ^{155}Eu capture cross-section correction.

6. Conclusion

This paper has described the CEA experimental program and associated results devoted to the validation of actinides and fission products build-up in PWR UOx spent fuel cycle. The qualification range extends up to 4.5% ^{235}U enrichment and high burnup fractions up to 60 GWd/t.

The depletion code DARWIN, based on the powerful code APOLLO2, has shown his capability to simulate the fuel inventory versus burnup. So, the DARWIN package is well suited for most of burnup credit nuclides inventory, except a slight underestimation of ^{241}Pu and overestimation of europium and gadolinium isotopes. However, the introduction in the future JEFF3.0 file of new evaluations, such as ^{241}Pu, $^{153\text{-}154}$Eu and ^{155}Eu will highly improved our C/E discrepancies.

An industrial package, named 'CIRACUSE', devoted to burnup credit studies, linking automatically DARWIN and CRISTAL, is in development; it will include correction factors applied to nuclide concentrations deduced from the experimental validation.

The P.I.E. data base is currently being extended to higher burnups with PWR rod cuts extracted after 5, 6 and 7 irradiation cycles (up to 80 GWd/t).

ACKNOWLEDGEMENT

The authors are indebted to Anne Barreau and Christine Chabert for their contribution to the P.I.E. calculations and to EDF/FRAMATOME/COGEMA for their financial support.

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