# Center for Nuclear Waste Regulatory Analyses

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December 12, 1996
Contract No. NRC-02-93-005
Project No. 20-5706-001

U.S. Nuclear Regulatory Commission
ATTN: Mr. Gary C. Comfort, Jr.
Office of Nuclear Material Safety & Safeguards
Mail Stop T8A-33
Washington, DC 20555

Subject: Letter Report Documenting the Review of the Safety Analysis Report (SAR) for the Fuel Receiving and Storage Facility, WVNS-SAR-012, Revision 0, Draft C, and Resolution of West Valley Demonstration Project (WVDP) Responses—Waste Solidification Systems Intermediate Milestone 20-5706-001-701

References:

1. Letter dated October 8, 1996, from T.J. Rowland (WVDP) to G. Comfort (NRC) transmitting the report entitled Safety Analysis Report for Fuel Receiving and Storage Facility, WVNS-SAR-012, Rev. 0, Draft C

2. Informal electronic mail dated October 22, 1996, from N. Sridhar (CNWRA) to G. Comfort (NRC) containing comments on WVNS-SAR-012, Rev. 0, Draft C

3. Letter dated November 5, 1996, from G. Comfort (NRC) to T.J. Rowland (WVDP) transmitting the comments on WVNS-SAR-012, Rev. 0, Draft C

4. Fax transmittal dated November 27, 1996, from J. Wolniewicz (Dames & Moore) to N. Sridhar (CNWRA) on responses to comments on WVNS-SAR-012, Rev. 0, Draft C

Dear Mr. Comfort:

Attachment 1 contains the comments generated by the CNWRA staff after reviewing the Safety Analysis Report (SAR) for the Fuel Receiving and Storage Facility, WVNS-SAR-012, Revision 0, Draft C. Also included in this attachment are the WVDP responses, provided by their contractor, Dames & Moore, and our resolution of these responses. The complete text and tables of the WVDP responses are provided as Attachment 2.

The WVDP responses have adequately resolved many of our comments and questions. However, three open items still remain. To resolve these open items, the following recommendations are made:

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426.1

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(i) Comment #2: Additional justification should be provided for the conclusion that the Fuel Receiving and Storage building meets design criteria.

(ii) Comment #6: Justification for the selection of key radionuclides listed in Tables 8.2-3 and 8.2-4 should be provided.

(iii) Comment #10: The dose calculations in the revised Table 9.2-3 should consider the presence of daughter products (e.g., Am-241) in the initial inventory in order to be conservative.

I trust that this transmittal meets your needs for responding to the WVDP on WVNS-SAR-012. If you need further clarification, please call me at (210) 522-5538.

Sincerely yours,

Narasi Sridhar

Narasi Sridhar
Element Manager
Engineered Barrier Systems and
Waste Solidification Systems

NS/blg
Attachment

cc: J. Linehan
S. Fortuna
B. Stiltenpole
B. Meehan
M. Tokar
R. Weller
K. Hardin
W. Patrick
CNWRA Directors
CNWRA Element Managers
S. Rowe (SwRI)

# ATTACHMENT 1

## REVIEW OF THE SAFETY ANALYSIS REPORT (SAR) FOR THE FUEL RECEIVING AND STORAGE FACILITY, WVNS-SAR-012, REV. 0, DRAFT C, AND RESOLUTION OF WEST VALLEY DEMONSTRATION PROJECT RESPONSES

## REVIEW OF THE SAFETY ANALYSIS REPORT (SAR) FOR THE FUEL RECEIVING AND STORAGE FACILITY, WVNS-SAR-012, REVISION 0, DRAFT C, AND RESOLUTION OF WEST VALLEY DEMONSTRATION PROJECT RESPONSES

The Center for Nuclear Waste Regulatory Analyses (CNWRA) performed a review of West Valley Nuclear Services, Inc. (WVNS) Safety Analysis Report (SAR) for Fuel Receiving and Storage (FRS) Facility, WVNS-SAR-012, Rev. 0, Draft C (SAR-012). The CNWRA submitted comments and questions on SAR-012 by electronic mail to the Nuclear Regulatory Commission (NRC) staff. Following discussions with CNWRA staff, the NRC transmitted modified comments to West Valley Demonstration Project (WVDP) staff. After receipt of the responses from the WVDP contractor, Dames & Moore, a further review of SAR-012 and the responses was conducted by the CNWRA staff. The review and resolution of the Dames & Moore responses by CNWRA staff is given below.

**Comment #1**

<u>§ 2, pages 2-4 and 2-6, lines 18-20 and Table 2.5-1</u>

> The text mentions that the dose to the maximally exposed individual off-site following a criticality event would be 399 millirem (mrem). This is inconsistent with the value given in Table 2.5-1 which indicates a dose of 967 mrem. It appears that the offsite and onsite doses have been reversed in this table. Please review this table to ensure correctness.

WVDP Response:

> The values corresponding to the on-site and off-site doses due to a criticality in the FRS were transposed in both Table 2.5-1 and Table 9.2-4. This accident was reevaluated in response to concerns raised in Comment #10. The corrected values have been incorporated into Tables 2.5-1 and 9.2-4 and into the text discussions of the accident in Chapters 2 and 9 of WVNS-SAR-012, Draft D.

CNWRA Evaluation of WVDP Response and Resolution:

> The new Tables 2.5-1 and 9.2-4 (Attachment 2) supplied by Dames & Moore indicate that the maximum off-site doses are less than the maximum on-site doses, which is internally consistent. However, the new dose values in these tables are significantly different from those mentioned in the original text of SAR-012, Rev. 0, Draft C. This is because the radionuclide inventory values were changed as a result of response to Comment #6. Hence, assuming that the new dose values are correct, the text in Sections 2 and 9 needs to be revised to match the new dose values provided in these tables. This item is considered to be resolved assuming that the text in Chapters 2 and 9 has been revised.

**Comment #2**

<u>§ 4.0 and 5.0, pages 4-1 and 5-3</u>

> The text states that although the FRS facility does not meet all of the current design criteria, it is nonetheless judged to meet the current needs of the West Valley Project. Please provide a copy of the Bixby (1989) reference that supports this assertion.

# REVIEW OF THE SAFETY ANALYSIS REPORT (SAR) FOR THE FUEL RECEIVING AND STORAGE FACILITY, WVNS-SAR-012, REVISION 0, DRAFT C, AND RESOLUTION OF WEST VALLEY DEMONSTRATION PROJECT RESPONSES

The Center for Nuclear Waste Regulatory Analyses (CNWRA) performed a review of West Valley Nuclear Services, Inc. (WVNS) Safety Analysis Report (SAR) for Fuel Receiving and Storage (FRS) Facility, WVNS-SAR-012, Rev. 0, Draft C (SAR-012). The CNWRA submitted comments and questions on SAR-012 by electronic mail to the Nuclear Regulatory Commission (NRC) staff. Following discussions with CNWRA staff, the NRC transmitted modified comments to West Valley Demonstration Project (WVDP) staff. After receipt of the responses from the WVDP contractor, Dames & Moore, a further review of SAR-012 and the responses was conducted by the CNWRA staff. The review and resolution of the Dames & Moore responses by CNWRA staff is given below.

**Comment #1**

<u>§ 2, pages 2-4 and 2-6, lines 18-20 and Table 2.5-1</u>

The text mentions that the dose to the maximally exposed individual off-site following a criticality event would be 399 millirem (mrem). This is inconsistent with the value given in Table 2.5-1 which indicates a dose of 967 mrem. It appears that the offsite and onsite doses have been reversed in this table. Please review this table to ensure correctness.

WVDP Response:

The values corresponding to the on-site and off-site doses due to a criticality in the FRS were transposed in both Table 2.5-1 and Table 9.2-4. This accident was reevaluated in response to concerns raised in Comment #10. The corrected values have been incorporated into Tables 2.5-1 and 9.2-4 and into the text discussions of the accident in Chapters 2 and 9 of WVNS-SAR-012, Draft D.

CNWRA Evaluation of WVDP Response and Resolution:

The new Tables 2.5-1 and 9.2-4 (Attachment 2) supplied by Dames & Moore indicate that the maximum off-site doses are less than the maximum on-site doses, which is internally consistent. However, the new dose values in these tables are significantly different from those mentioned in the original text of SAR-012, Rev. 0, Draft C. This is because the radionuclide inventory values were changed as a result of response to Comment #6. Hence, assuming that the new dose values are correct, the text in Sections 2 and 9 needs to be revised to match the new dose values provided in these tables. This item is considered to be resolved assuming that the text in Chapters 2 and 9 has been revised.

**Comment #2**

<u>§ 4.0 and 5.0, pages 4-1 and 5-3</u>

The text states that although the FRS facility does not meet all of the current design criteria, it is nonetheless judged to meet the current needs of the West Valley Project. Please provide a copy of the Bixby (1989) reference that supports this assertion.

WVDP Response:

A copy of Bixby (1989) has been provided to the reviewer.

CNWRA Evaluation of WVDP Response and Resolution:

The Bixby (1989) reference does not provide technical information relevant to this comment; however, the Dames & Moore (1995) report does. A three-dimensional (3D) finite element dynamic analysis of the FRS and process buildings, including soil-structure interaction modeling, conducted by Dames & Moore, shows that the lateral load-resisting bracing of the FRS steel frame yields or buckles at a peak ground acceleration (PGA) of 0.05 g [0.5 × Evaluation Basis Earthquake (EBE)] in the east-west direction and at a PGA of 0.075 g (0.75 × EBE) in the north-south direction. However, despite yielding or buckling of the lateral load-resisting bracing of the FRS steel frame at 50 percent to 75 percent of EBE, Dames & Moore concluded, without any quantitative justification, that significantly higher levels of ground motion would be required to induce failures leading to collapse of the building. While the entire FRS building may not collapse, a large section of it might and could pose a threat to the spent nuclear fuel assemblies stored in the FRS pool. The WVDP should provide quantitative justification to support its conclusion regarding the stability of this building.

**Comment #3**

<u>§ 5.2.3, pages 5-4 and 5-5</u>

The text states that masonry blocks from the 8-inch masonry block interface wall between the FRS building and the Main Plant building may become dislodged during certain seismic conditions. Please provide a copy of the Dames & Moore (1995) reference evaluating this issue.

WVDP Response:

A copy of the Dames & Moore (1995) report has been provided to the reviewer.

CNWRA Evaluation of WVDP Response and Resolution:

The Dames & Moore (1995) report evaluated the risk of potential collapse of the 8-inch masonry block wall adjacent to the FRS spent fuel pool under extreme natural hazard seismic conditions. The Dames & Moore activities included a review of the findings of previous independent seismic integrity studies by Blaw-Knox (1972), Dravo (1976), Lawrence Livermore National Laboratory (LLNL) (1978), and Los Alamos National Laboratory (LANL) (1978), as well as a 3D finite element dynamic analysis of the FRS and process buildings, including soil-structure interaction modeling.

The Dames & Moore review of the previous independent seismic integrity studies indicated that the estimate of the load-bearing capacity of the 8-inch masonry block wall adjacent to the pool was based on inconsistent data. To correct this deficiency, Dames & Moore performed a 3D finite element dynamic analysis to more accurately assess the wall's performance under various levels of earthquake loading. The assumptions made by Dames & Moore in conducting this dynamic analysis are consistent with standard engineering practice. Based on the dynamic

analysis, the 8-inch masonry block wall does not pose a threat to the spent nuclear fuel assemblies stored in the FRS pool for an EBE of 0.1 g PGA. The issue of the risk of potential collapse of this masonry block wall under extreme natural hazard seismic conditions is considered to be resolved.

**Comment #4**

Page 6.8, line 19

The meaning of water quality in this section is unclear. It may refer to the decontamination factor of the demineralizer system or to different water properties such as pH, conductivity, etc. It appears that the measurement of $\beta$ activity or $^{137}Cs$ $\gamma$ activity is the method used for detection of fuel failures, but this is not clear from the text. Please describe the water quality requirements that are evaluated and clarify the method(s) of fuel failure detection.

WVDP Response:

The text will be modified to indicate that water samples from the fuel storage pool or CUP will be collected weekly to measure pH, conductivity, $^{137}Cs$ $\gamma$ activity, and gross $\beta$ activity. These parameters will also be measured quarterly as well as chlorides, nitrates, nitrites, sulfates, and gross $\alpha$ activity. It is stated that the purpose of this sampling and analysis is to verify the performance of the fuel pool Submerged Water Filtration System in maintaining water quality parameters within desired ranges and to detect possible fuel failures.

CNWRA Evaluation of WVDP Response and Resolution:

Based on the WVDP response, this comment is considered to be resolved. However, the maximum acceptable values for the various parameters in the form of a table, and the recommended corrective actions for upset conditions should be provided.

**Comment #5**

Table 7.2.1

Some of the information presented in this table does not appear to be internally consistent. Examples include: (1) the total Pu in HIC "C" is less than the sum of the Pu isotopes in the same HIC, and (2) the total $^{238}Pu$ inventory assuming the indicated sludge mass for all HICs is 0.0116 rather than 0.0193 Ci. Values in the table should be reviewed to ensure correctness.

WVDP Response:

The value for Pu-239/240 activity concentration for HIC "C" will be corrected. An additional correction to the table will be the elimination of total activity values for Co-60, Pu-238 and Pu-239/240 because activity concentrations for these isotopes were not measured for all HICs.

CNWRA Evaluation of WVDP Response and Resolution:

Based on the WVDP response, this comment is considered to be resolved.

**Comment #6**

Tables 8.2-3 and 8.2-4 on page 8-28 and 8-29, respectively

The 21 yr PWR and BWR fuel inventories for Am-241 are less than their respective initial inventories. Due to ingrowth from the decay of Pu-241, the inventory of Am-241 increases with time (for short times). Also, the basis for choosing the presented radionuclides as the "key radionuclides" for this exercise has not been referenced nor explained.

An independent analysis using ORIGEN version 2.1 found that for the PWR fuel with the listed characteristics, the initial inventory for Am-241 was 133 Ci/MTU (a 50% increase from the 86 Ci/MTU listed in Table 8.2-4). The 21-yr inventory of Am-241 was found to be 2,673 Ci/MTU (a factor of 32 increase from the 83 Ci/MTU listed in the table). For all other nuclides, independent analysis using ORIGEN version 2.1 roughly agreed with the results listed in the table. It appears that the authors of the report are relying on ORIGEN runs that were performed for the initial fuel inventories and then are calculating the 21-yr inventories by correcting for only the decay of the nuclide. This process is incorrect for nuclides that appear in a decay chain, such as Am-241.

For the BWR fuel, the transuranics listed in Table 8.3-3, with the exception of Pu-238, were a factor of 3 to 5 lower than predicted by the independent analysis. Specifically for Am-241, the initial inventory was found to be 25 Ci/MTU (a factor of 5 increase from the 5 Ci/MTU listed in Table 8.2-3) for a fuel with an initial enrichment of 2.75% U-235 exposed to a specific power of 25.9 kw/kg with a burnup of 12,423 Mwd/MTU. The initial enrichment was assumed since none was listed in the table. The 21-yr inventory of Am-241 was found to be 85 Ci/MTU (a factor of 18 increase from the 4.83 Ci/MTU value listed in the table). Again, it appears that the authors of the report are relying on ORIGEN runs that were performed for the initial fuel inventories and then are calculating the 21-yr inventories by correcting for only the decay of the nuclide.

The differences in the inventory may affect subsequent calculations, especially any criticality calculations, that use Tables 8.2-3 and 8.2-4 as a basis for the radionuclide content of the fuel.

WVDP Response:

The values for Am-241 activity in Tables 8.2-3 and 8.2-4 of SAR-012, Draft C, were incorrectly calculated. Correct values have been calculated for both BWR and PWR fuel using ORIGEN2, and these values are included in the appropriate tables in SAR-012. The revised Am-241 activities to be included in the SAR are based on the more realistic decay times of 22 and 24 years for the BWR and PWR fuel, respectively.

CNWRA Evaluation of WVDP Response and Resolution:

The recalculated values of radionuclide inventory shown in revised Tables 8.2-3 and 8.2-4 (Attachment 2) appear reasonable for all radionuclides (including Am-241). Therefore, this portion of the response is accepted. However, the authors have not justified the selection of the radionuclides shown in Tables 8.2-3 and 8.2-4 as the "important" radionuclides for this analysis. For example, the initial inventories of Am-243 and Cm-243 may be about equal, and

they have approximately equal dose conversion factors. However, one nuclide is tracked (Am-241) and the other is not (Cm-243) in the analysis presented in SAR-012. Other actinides, such as Am-242m, may also be important.

**Comment #7**

Tables 8.7-1 and 8.7-2 on page 8-30 and 8-31, respectively

Neither the text nor the subject tables state whether burnup credit was considered when calculating the listed values of $k_{eff}$. If the uranium and plutonium inventories shown in Tables 8.2-1 and 8.2-2 were used for the criticality calculations, that should also be stated.

WVDP Response:

All criticality analyses referenced in the SAR have assumed unirradiated fuel, unless explicitly stated otherwise.

The following text has been added after the sentence ending on Line 1 of page 8-12: "All calculations were performed assuming unirradiated fuel."

The sentence beginning in Line 10 on page 8-12 has been rev orded as follows: "This analysis evaluated the reactivity of unirradiated PWR and BWR assemblies . . ."

CNWRA Evaluation of WVDP Response and Resolution:

Since the "fresh fuel" assumption was used in performing the criticality calculations, changes in the radionuclide inventory due to irradiation and decay would not affect the calculated criticality calculations. For fresh fuel, the calculated values of $k_{eff}$ shown in Tables 8.7-1 and 8.7-2 appear reasonable. On the basis of the WVDP response, this comment is considered to be resolved.

**Comment #8**

§ 9, page 9-3, line 25 and § 9.1.2.3.2

The text states that ". . . gravitational potential energy represents the most significant source of energy . . ." Have the consequences of combustion (i.e., fire hazards) been considered in making this evaluation? It would seem that fire would have the potential for significant radiological and nonradiological impacts.

WVDP Response:

The consequences of a fire in the FRS were evaluated in the process hazards analysis and have been documented in Table 9.1-1. A fire in the FRS is not considered to pose a significant risk because the FRS contains only minor amounts of flammable materials. Furthermore, hazards located in the FRS (i.e., spent nuclear fuel, pool water filtration cartridges, and loaded ion exchange resin) are stored either underwater or in large concrete shield containers, which would provide protection for the hazards from a fire, should one occur.

CNWRA Evaluation of WVDP Response and Resolution:

Based on the WVDP response, this comment is considered to be resolved.

**Comment #9**

§ 9, page 9-24, Table 9.2-1

Due to discrepancies found in the inventory calculations, the on-site dose calculation for Am-241 for a Class D atmosphere was checked. The listed Chi/Q value was reasonably well reproduced ($1.5E-06$ s/m$^3$ versus $1.6E-06$ listed in the SAR) but because of the difference in the calculated inventory of Am-241 used in the SAR, the on-site dose value in the table was calculated to be $5.8E-05$ rem (the same factor of 32 increase found in Comment #6). These results make Am-241 the top-ranking nuclide in terms of dose in the list.

WVDP Response:

A reanalysis of the consequences of the failure of the fuel assemblies in the FRS has been performed using the corrected values of Am-241. Results of this analysis are documented in Table 9.2-1 of SAR-012, Draft D. The dose to the maximally exposed off-site individual from the drop of a single fuel assembly is $6.14E-03$ rem while the dose due to the failure of all 125 fuel assemblies has been calculated to be $7.68E-01$ rem.

CNWRA Evaluation of WVDP Response and Resolution:

Table 9.2-1 of SAR-012, Draft D (Attachment 2), was reviewed. In this revised table, the authors used corrected inventory values for all nuclides (calculated using the most recent version of ORIGEN2) as well as for Am-241. This strategy is appropriate. The new values appear to be reasonable. Therefore, this comment is considered to be resolved.

**Comment #10**

§ 9, page 9-13, lines 30 and 31

Due to possible errors in the radionuclide inventories listed in Tables 8.2-3 and 8.2-4, these analyses should be redone with correct inventories (if the original inventories are determined to be in error).

WVDP Response:

A reanalysis of the consequences of an inadvertent criticality in the FRS has been performed using the activities presented in Table 8.2-4. Results of this analysis are documented in Table 9.2-3 of SAR-012, Draft D. The dose to the maximally exposed off-site individual from this event is calculated as $3.53E-01$ rem.

CNWRA Evaluation of WVDP Response and Resolution:

In revised Table 9.2-3 of SAR-012, Draft D (Attachment 2), the authors have used the initial inventory for all radionuclides in the dose calculations. For nuclides that do not appear in a decay chain, assuming the initial isotopic content of the fuel yields conservative dose calculations. For nuclides appearing in a decay chain (such as Am-241), this assumption can be nonconservative. It is recommended that the authors assume the 24-yr inventory for Am-241 when calculating the doses shown in Table 9.2-3. Using the 24-yr inventory for Am-241 will increase its importance by approximately a factor of 26, placing it fourth in Table 9.2-3.

## REFERENCES


Bixby, Willis W. 1989. DOE Order 6430. 1A. *Letter dated July 17, 1989, to R.A. Thomas.* CBL:010:89-0902:89:01 (DW:89:0365).

Blaw-Knox. 1972. *Evaluation of the Fuel Receiving and Storage Structure for Tornado and Earthquake Forces.*

Dames & Moore. 1995. *Seismic Integrity Review Report—Fuel Receiving and Storage Facility.* West Valley, NY: Dames & Moore.

Dravo. 1976. *Seismic Competence of the Existing Reprocessing Building at the West Valley Reprocessing Plant.* Chemical Plants Division. Report No. 0476.015. West Valley, NY.

Lawrence Livermore National Laboratory. 1978. *Structural Analyses of the Fuel Receiving Station Pool at the Nuclear Fuel Service Reprocessing Plant.* UCRL-52572. Livermore, CA: Lawrence Livermore National Laboratory.

Los Alamos National Laboratory. 1978. *Seismic Investigation of the Nuclear Fuel Services, Inc. Reprocessing Plant at West Valley, NY.* LA-7087-MS. Los Alamos, NM: Los Alamos National Laboratory.

# ATTACHMENT 2

## WEST VALLEY DEMONSTRATION PROJECT RESPONSE TO NUCLEAR REGULATORY COMMISSION COMMENTS ON WVNS-SAR-012, REV. 0, DRAFT C

**Department of Energy**
Ohio Field Office
West Valley Area Office
P.O. Box 191
West Valley, NY 14171

December 5, 1996

Mr. Gary C. Comfort
U.S. Nuclear Regulatory Commission
MS T8D14
Washington, DC 20555

SUBJECT: Response to U.S. Nuclear Regulatory Commission (NRC) Comments on WVNS-SAR-012, Rev. 0, Draft C (TAC NO. L30924)

REFERENCES: 1) Letter 0554:95:08, G. C. Comfort to T. J. Rowland, "Comments on the Safety Analysis Report (SAR) for Fuel Receiving and Storage Facility, WVNS-SAR-012, Revision 0, Draft C," dated November 5, 1996

2) Letter WD:96:0915 (0554:95:08), T. G. Weiss to T. J. Rowland, "Responses to NRC Comments on WVNS-SAR-012, Rev. 0, Draft C," dated December 4, 1996

Dear Mr. Comfort:

In Reference 1, the NRC provided the DOE West Valley Demonstration Project (DOE-WV) with NRC's comments on WVNS-SAR-012, Rev. 0, Draft C, entitled "Safety Analysis Report for Fuel Receiving and Storage Facility." Draft responses to your comments were previously provided to you. Based upon follow-up discussions with you, the West Valley Demonstration Project has finalized our responses. The responses are provided in Reference 2, which is enclosed.

As previously discussed with you, DOE-WV anticipated receiving a letter report from the NRC documenting your safety evaluation. This letter will be used as supporting documentation in the DOE Safety Evaluation Report.

If you have any questions regarding this matter, please contact Bryan Bower at (716) 942-4368.

Sincerely

Richard E. Loucher for
T. J. Rowland, Director
West Valley Demonstration Project

Enclosure: Reference 2

cc: N. Sridhar, SRI, w/enc.
T. G. Weiss, WVNS, WV-AA9, w/o enc.

BCB:080:96 - 0554:95:08
BCB/jam

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West Valley Nuclear Services Company, Inc

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Bryan

T. J. Rowland, Director
DOE West Valley Demonstration Project
P.O. Box 191
West Valley, New York 14171-0191

WV-AA9
WD:96:0915
December 4, 1996

ATTENTION: T. J. Jackson

RESPONSE
DW:7057

Dear Mr. Rowland:

SUBJECT: Responses to NRC comments on WVNS-SAR-012, Rev.0, Draft C

REFERENCES: 1) DW:96:0958 (BCB:076:96-0554:95:08), R.B. Provencher to W.G. Poulson, "U.S. Nuclear Regulatory Commission (NRC) Comments on WVNS SAR-012", dated November 15, 1996.

2) NRC letter, Gary Comfort, Jr. to T.J. Rowland, "Comments on the Safety Analysis Report (SAR) for the Fuel Receiving and Storage Facility, WVNS-SAR-012, Revision 0, Draft C", dated November 5, 1996.

Attached, as requested in Reference 1, are responses to NRC comments on WVNS-SAR-012, Rev.0, Draft C, which were transmitted to DOE-WV by Reference 2.

If there are any questions, please contact the undersigned at extension 4668. This transmittal completes action DW:7057.

Very truly yours,
West Valley Nuclear Services Co., Inc.

Thomas G. Weiss, Jr., Manager,
Site Projects

EN:96:0075

TGW:klm

Attachment: Responses to NRC Comments on WVNS-SAR-012

cc: B. C. Bower, DOE-WV Area Office, WV-DOE
H. E. Moore, DOE-WV Area Office, WV-DOE

3127klm



C554 95:17

**West Valley Demonstration Project**
**Response to NRC Comments on WVNS-SAR-012**

**Comment #1**

<u>Section 2, Page 2-4 (Lines 18-20 and Table 2.5-1)</u>

The text mentions that the dose to the maximally exposed individual off-site following a criticality event would be 399 millirem (mrem). This is inconsistent with the value given in Table 2.5-1 which indicates a dose of 967 mrem. It appears that the offsite and onsite doses have been reversed in this table. Please review this table to ensure its correctness.

**Response**

The values corresponding to the on-site and off-site doses due to a criticality in the FRS have been transposed in both Table 2.5-1 and Table 9.2-4. The values as stated in the text are correct. Tables 2.5-1 and 9.2-4 have been corrected to reflect the values given in the text.

**Comment #2**

<u>Sections 4.0 and 5.0, Pages 4-1 and 5-3</u>

The text states that although the Fuel Receiving and Storage (FRS) facility does not meet all of the current design criteria, they are nonetheless judged to meet the West Valley Project's current needs. Please provide a copy of the Bixby (1989) reference that supports this assertion.

**Response**

A copy of Bixby, 1989 has been provided to the reviewer.

**Comment #3**

<u>Section 5.2.3, Pages 5-4 to 5-5</u>

The text states that masonry blocks from the 8-inch masonry block interface wall between the FRS building and the Main Plant building may become dislodged during certain seismic conditions. Please provide a copy of the Dames & Moore (1995) reference evaluating this issue.

**Response**

A copy of the Dames & Moore, 1995 report has been provided to the reviewer.

**Comment #4**

Page 6.8, Line 19

The meaning of water quality in this section is unclear. It may refer to the decontamination factor of the demineralizer system or to different water properties such as pH, conductivity, etc. It appears that the measurement of Beta activity or Cs-137 gamma activity is the method used for detection of fuel failures, but this is not clear from the text. Please describe the water quality requirements that are evaluated and clarify the method(s) of fuel-failure detection.

**Response**

The first two sentences beginning at line 17 on page 6-8 has been replaced by the following: "Water samples from the fuel storage pool or CUP are collected weekly and analyzed for the water quality parameters of pH, conductivity, Cs-137 gamma activity, and gross beta activity. Quarterly samples are analyzed for these parameters as well as chlorides, nitrates, nitrites, sulfates, and gross alpha activity. The purpose of this sampling and analysis is to verify the performance of the fuel pool Submerged Water Filtration System in maintaining water quality parameters within desired ranges. Additionally, possible fuel failures can be detected by Cs-137 gamma activity and gross beta activity measurements."

**Comment #5**

Table 7.2-1

Some of the information presented in this table does not appear to be internally consistent. Examples include: (1) the total Pu in HIC "C" is less than the sum of the Pu isotopes in the same HIC, and (2) the total Pu-238 inventory assuming the indicated sludge mass for all HICs is 0.0116 rather than 0.0193 Ci. Please review these tables to ensure their correctness.

**Response**

As indicated in the note at the bottom of Table 7.2-1, concentration values given in the table are an average of concentrations of activities in samples collected from the top, middle, and bottom of the respective HIC's. The Pu-239/240 activity concentration given in the table for HIC "C", however, is the total activity for these samples. The correct average value of 1.74E-2 μCi/g will replace the value of 5.22E-2 μCi/g stated in the table.

The concentration of Co-60 activity was not determined for HIC "A" and isotopic measurements of Pu-238 and Pu-239/240 were not made for HICs "A" and "B". For the purposes of calculating the total activity in resin waste, estimates of these values were made based on the ratio of the unmeasured parameter to other measured isotopes. The following text has been added to Note #6 of Table 7.2-1: "Total Co-60 activity based on an estimated concentration of Co-60 in HIC 'A'. Estimate determined from Cs-137 concentration in HIC 'A' and average of Co-60:Cs-137 ratios in HICs 'B' through 'E'. Total Pu-238 and Pu-239/240 activities based on estimated concentrations of Pu-238 and Pu-239/240 in HICs 'A' and 'B'. Estimates determined from Total Pu concentration in respective HIC (A or B) and average of Pu isotopic:Total Pu ratios in HICs 'C' through 'E'."

**Comment #6**

Tables 8.2-3 and 8.2-4 on Pages 8-28 and 8-29, respectively

The 21-yr PWR and BWR fuel inventories for Am-241 are less than their respective initial inventories. Due to ingrowth from the decay of Pu-241, the inventory of Am-241 is expected to increase with time (for short times). This discrepancy may be a result of the initial inventories being calculated with an old version of ORIGEN and then calculating the 21-year inventories without accounting for ingrowth resulting from the decay chain. Also, the basis for choosing the presented radionuclides as the "key radionuclides" for this exercise has not been referenced nor explained.

An independent analysis using ORIGEN Version 2.1 found that for the PWR fuel with the listed characteristics, the initial inventory for Am-241 was 133 Ci/MTU (a 50% increase form the 86 Ci/MTU listed in Table 8.2-4). The 21-yr inventory of Am-241 was found to be 2,673 Ci/MTU (a factor of 32 increase from the 83 Ci/MTU listed in the table). For all other nuclides, independent analysis using ORIGEN Version 2.1 roughly agreed with the results listed in the table. It appears that the authors of the report are relying on ORIGEN runs that were performed for the initial fuel inventories and then are calculating the 21-year inventories by correcting for only the decay of the nuclide. This process is incorrect for nuclides that appear in a decay chain, such as Am-241.

For the BWR fuel, the transuranic listed in Table 8.3-3, with the exception of Pu-238, were a factor of 3 to 5 lower than predicted by the independent analysis. Specifically for Am-241, the initial inventory was found to be 25 Ci/MTU (a factor of 5 increase from the 5 Ci/MTU listed in Table 8.2-3) for a fuel with an initial enrichment of 2.75% U-235 exposed to a specific power of 25.9 kw/kg with a burnup of 12,423 Mwd/MTU. The initial enrichment was assumed since none was listed in the table. The 21-yr inventory of Am-241 was found to be 85 Ci/MTU (a factor of 18 increase from the 4.83 Ci/MTU value listed in the table). Again, it appears that the authors of the report are relying on ORIGEN runs that were performed for the initial fuel inventories and then are calculating the 21-yr inventories by correcting for only the decay of the nuclide.

The differences in the inventory may affect subsequent calculations, especially any critically calculations, that use Tables 8.2-3 and 8.2-4 as a basis for the radionuclide content of the fuel.

**Response**

The values for Am-241 activity in Tables 8.2-3 and 8.2-4 of SAR-012, Draft C, were incorrectly calculated. Correct values have been calculated for both BWR and PWR fuel using ORIGEN2 and these values are included in the appropriate tables in SAR-012. The revised Am-241 activities to be included in the SAR are based on the more realistic decay times of 22 years and 24 years for the BWR and PWR fuel, respectively.

**Comment #7**

Tables 8.7-1 and 8.7-2 on Pages 8-10 and 8-11, respectively

The text nor the subject tables state whether burnup credit was considered when calculating the listed values of $k_{eff}$. If the uranium and plutonium inventories shown in Tables 8.2-1 and 8.2-2 were used for the criticality calculations, that should also be stated.

**Response**

All criticality analyses referenced in the SAR have assumed unirradiated fuel unless explicitly stated otherwise.

The following text has been added after the sentence ending on Line 1 of page 8-12: "All calculations were performed assuming unirradiated fuel."

The sentence beginning in Line 10 on page 8-12 has been reworded as follows: "This analysis evaluated the reactivity of unirradiated PWR and BWR assemblies, . . . ."

**Comment #8**

Section 9, Page 9-3, Line 25 and Section 9.1.2.3.2

The text states that "....gravitational potential energy represents the most significant source of energy..." Have the consequence of combustion (i.e., Fire Hazards) been considered in making this evaluation? It would seem that fire would have the potential for significant radiological and non-radiological impacts.

**Response**

The consequences of a fire in the FRS were evaluated in the process hazards analysis and have been documented in Table 9.1-1. A fire in the FRS is not considered to pose a significant risk because the FRS contains only minor amounts of flammable materials. Furthermore, hazards located in the FRS (i.e., spent nuclear fuel, pool water filtration cartridges, and loaded ion exchange resin) are stored either underwater or in large concrete shield containers, which would provide protection for the hazards from a fire, should one occur.

**Comment #9**

Section 9, Page 9-24, Table 9.2-1

Due to discrepancies found in the inventory calculations, the on-site dose calculation for Am-241 for a class D atmosphere was checked. The listed Chi Q value were reasonably well reproduced ($1.5E-06$ s/m$^3$ versus $1.6E-06$ listed in the SAR) but because of the difference in the calculated inventory of Am-241 used in the SAR, the on-site dose value in the table was calculated to be 5.8E-05 rem (the same factor of 32 increase found in Comment #6). These results make Am-241 the top ranking nuclide in terms of dose in the list.

**Response**

A reanalysis of the consequences of the failure of the fuel assemblies in the FRS has been performed using the corrected values of Am-241. Results of this analysis are documented in Table 9.2-1 of SAR-012, Draft D. The dose to the maximally exposed off-site individual from the drop of a single fuel assembly is 6.14E-03 rem while the dose due to the failure of all 125 fuel assemblies has been calculated to be 7.68E-01 rem.

**Comment #10**

Section 9, Page 9-13, Lines 30 and 31

Due to possible errors in the radionuclide inventories listed in Tables 8.2-3 and 8.2-4, these analyses should be redone with correct inventories (if the original inventories are determined to be incorrect).

**Response**

A reanalysis of the consequences of an inadvertent criticality in the FRS has been performed using the activities presented in Table 8.2-4. Results of this analysis are documented in Table 9.2-3 of SAR-012, Draft D. The dose to the maximally exposed off-site individual from this event is calculated as 3.53E-01 rem.

TABLE 2.5-1

SUMMARY OF FRS ACCIDENT CONSEQUENCES

| Accident Scenario | Maximum Off-Site Dose (rem) | Maximum On-Site Dose (rem) | Evaluation Guideline Level |
|---|---|---|---|
| Dropping of a Fuel Assembly in the FRS | 6.14E-03 | 1.49E-02 | On-site - 25 rem |
| | | | Off-site - 5 rem |
| Dropping of a Loaded High Integrity Container | 2.74E-03 | 5.77E-03 | On-site - 100 rem |
| | | | Off-site - 25 rem |
| Inadvertent Criticality in the FRS | 3.53E-01 | 8.57E-01 | On-site - 100 rem |
| | | | Off-site - 25 rem |
| Failure of 125 SNF Assemblies Due to Seismic Event | 7.68E-01 | 1.86E+00 | On-site - Natural Phenomena, N/A |
| | | | Off-site - 25 rem |

TABLE 9.2-4

SUMMARY OF FRS ACCIDENT CONSEQUENCES

| Accident Scenario | Maximum Off-Site Dose (rem) | Maximum On-Site Dose (rem) | Evaluation Guideline Level |
|---|---|---|---|
| Dropping of a Fuel Assembly in the FRS | 6.14E-03 | 1.49E-02 | On-site - 25 rem |
| | | | Off-site - 5 rem |
| Dropping of a Loaded High Integrity Container | 2.74E-03 | 5.77E-03 | On-site - 100 rem |
| | | | Off-site - 25 rem |
| Inadvertent Criticality in the FRS | 3.53E-01 | 8.57E-01 | On-site - 100 rem |
| | | | Off-site - 25 rem |
| Failure of 125 SNF Assemblies Due to Seismic Event | 7.68E-01 | 1.86E+00 | On-site - Natural Phenomena, N/A |
| | | | Off-site - 25 rem |

**TABLE 7.2-1**

**RADIOLOGICAL CHARACTERISTICS OF HIC CONTENTS**

| Parameter | HIC "A"[1] (µCi/g) | HIC "B"[2] (µCi/g) | HIC "C"[3] (µCi/g) | HIC "D"[4] (µCi/g) | HIC "E"[5] (µCi/g) | Total Activity[6] (Ci) |
|---|---|---|---|---|---|---|
| Gross α | 6.65E-02 | 3.06E-01 | 3.49E-02 | 4.10E-02 | 6.00E-02 | 1.63E+00 |
| Gross β | 3.37E+01 | 8.10E+01 | 3.03E+01 | 4.24E+01 | 2.67E+01 | 6.85E+02 |
| Co-60 | ----- | 1.63E+00 | 8.37E-02 | 1.26E-02 | 3.42E-03 | 6.03E+00 |
| Sr-90 | 1.43E+00 | 6.85E-01 | 3.36E-01 | 4.32E-01 | 7.16E-02 | 9.44E+00 |
| Cs-137 | 2.24E+01 | 6.52E+01 | 2.97E+01 | 2.92E+01 | 2.56E+01 | 5.51E+02 |
| Pu-238 | ----- | ----- | 3.13E-03 | 3.95E-04 | 9.12E-05 | 9.11E-02 |
| Pu-239/ 240 | ----- | ----- | 1.74E-02 | 2.88E-03 | 6.00E-04 | 5.77E-01 |
| Am-241 | 2.95E-02 | 5.63E-02 | 1.56E-02 | 3.63E-03 | 1.29E-02 | 3.78E-01 |
| Total Pu | 8.70E-02 | 9.69E-02 | 2.05E-02 | 3.26E-03 | 6.90E-04 | 6.67E-01 |

Notes:

Concentrations based on average of top, middle, bottom samples.

(1) - Ref: Analytical Request Form No. 87-3131.

(2) - Based on average of top, middle, bottom samples, and analysis from separate sample. Ref: Analytical Request Form Nos. 86-1730 & 87-1113.

(3) - Ref: Analytical Request Form No. 9101655.

(4) - Ref: Analytical Request Form No. 9203040 (as amended by memo# IH:93:0044 [WVNS, 1993]).

(5) - Ref: Analytical Request Form No. 96-0485.

(6) - Based on the sum of activity from all HICs and an assumed mass of sludge of 3.2E+6 g per HIC. Total Co-60 activity based on an estimated concentration of Co-60 in HIC "A". Estimate determined from Cs-137 concentration in HIC "A" and average of Co-60:Cs-137 ratios in HICs "B" through "E". Total Pu-238 and Pu-239/240 activities based on estimated concentrations of Pu-238 and Pu-239/240 in HICs "A" and "B". Estimates determined from Total Pu concentration in respective HIC (A or B) and average of Pu isotopic:Total Pu ratios in HICs "C" through "E".

**TABLE 8.2-3**

**RADIOACTIVE CHARACTERISTICS OF BWR FUEL[1]**
**(PER METRIC TON U CHARGED TO REACTOR)**

| **Nuclide** | **Initial Fuel Activity[2] (Ci)** | **Decayed Fuel Activity[3] (Ci)** |
|---|---|---|
| Fe-55 | 1.72E+03 | 4.89E+00 |
| Co-60 | 5.11E+03 | 2.83E+02 |
| Ni-63 | 4.39E+02 | 3.72E+02 |
| Kr-85 | 5.75E+03 | 1.39E+03 |
| Sr-90 | 4.57E+04 | 2.70E+04 |
| Y-90 | 4.81E+04 | 2.71E+04 |
| Ru-106 | 1.37E+05 | 3.87E-02 |
| Sb-125 | 5.44E+03 | 2.24E+01 |
| Cs-134 | 5.94E+04 | 3.65E+01 |
| Cs-137 | 5.12E+04 | 3.08E+04 |
| Ba-137m | 4.85E+04 | 2.91E+04 |
| Pm-147 | 8.92E+04 | 2.83E+02 |
| Sm-151 | 1.76E+02 | 1.52E+02 |
| Eu-154 | 3.18E+03 | 5.39E+02 |
| Eu-155 | 1.86E+03 | 8.60E+01 |
| Pu-238 | 5.62E+02 | 4.94E+02 |
| Pu-239 | 7.50E+01 | 7.66E+01 |
| Pu-240 | 1.04E+02 | 1.04E+02 |
| Pu-241 | 1.98E+04 | 6.86E+03 |
| Am-241 | 1.04E+01 | 4.31E+02 |
| Cm-244 | 5.67E+01 | 2.45E+01 |

Notes:

[1] BWR Fuel: 3.0 w/o U-235; burnup 16,111 MWD/MTU; specific power - 25.9 MW/MTU

[2] Isotopic content of fuel at reactor discharge

[3] Isotopic content of fuel decayed 22 years

TABLE 8.2-4

RADIOACTIVE CHARACTERISTICS OF PWR FUEL[1]
(PER METRIC TON U CHARGED TO REACTOR)

| Nuclide | Initial Fuel Activity[2] (Ci) | Decayed Fuel Activity[3] (Ci) |
|---|---|---|
| H-3 | 8.00E+02 | 2.08E+02 |
| Fe-55 | 5.28E+03 | 8.80E+00 |
| Co-60 | 7.49E+03 | 3.19E+02 |
| Ni-63 | 6.60E+02 | 5.51E+02 |
| Kr-85 | 9.49E+03 | 2.01E+03 |
| Sr-90 | 7.47E+04 | 4.22E+04 |
| Y-90 | 8.06E+04 | 4.22E+04 |
| Tc-99 | 1.31E+01 | 1.31E+01 |
| Ru-106 | 5.00E+05 | 3.40E-02 |
| Sb-125 | 1.49E+04 | 3.71E+01 |
| I-129 | 3.07E-02 | 3.10E-02 |
| Cs-134 | 1.45E+05 | 4.54E+01 |
| Cs-137 | 1.03E+05 | 5.94E+04 |
| Ba-137m | 9.81E+04 | 5.62E+04 |
| Pm-147 | 1.30E+05 | 2.43E+02 |
| Sm-151 | 3.52E+02 | 2.99E+02 |
| Eu-154 | 9.95E+03 | 1.44E+03 |
| Eu-155 | 6.12E+03 | 2.14E+02 |
| Pu-238 | 2.12E+03 | 1.96E+03 |
| Pu-239 | 3.05E+02 | 3.11E+02 |
| Pu-240 | 5.10E+02 | 5.11E+02 |
| Pu-241 | 1.19E+05 | 3.74E+04 |
| Am-241 | 1.07E+02 | 2.75E+03 |
| Am-243 | 1.42E+01 | 1.42E+01 |
| Cm-244 | 1.50E+03 | 5.99E+02 |

Notes:

[1] PWR Fuel: 3.3 w/o U-235; burnup 33,000 MWD/MTU; specific power - 30 MW/MTU

[2] Isotopic content of fuel at reactor discharge

[3] Isotopic content of fuel decayed 24 years

## TABLE 9.2-1

## DROP OF A FUEL ASSEMBLY IN THE FRS

Assumptions:
ARF x RF x DR x LPF (Pool Release) 1.05E-04 - See Note [1] Below
ARF x RF x DR x LPF (Atmospheric Release) 2.00E-03 - See Note [1] Below
Release Height 60 m (Elevated Release)

| Receptor Location | | | | 640 m | 640 m | 640 m | 1050 m | 1050 m | 1700 m | |
|---|---|---|---|---|---|---|---|---|---|---|
| Stability Class, Wind Speed | | | | D, 4.5 m/s | F, 1 m/s | 95% | D, 4.5 m/s | F, 1 m/s | 95% | |
| Dispersion (χ/Q) | | | | 1.59E-06 $s/m^3$ | 2.70E-11 $s/m^3$ | 1.63E-04 $s/m^3$ | 5.54E-06 $s/m^3$ | 1.03E-07 $s/m^3$ | 6.72E-05 $s/m^3$ | |
| Nuclide | Decayed PWR Activity (Ci) | Pool Source Term (Ci) | Atmospheric Source Term (Ci) | On-Site Dose (rem) | On-Site Dose (rem) | On-Site Dose (rem) | Off-Site Dose (rem) | Off-Site Dose (rem) | Off-Site Dose (rem) | Percent Dose Contribution |
| Am-241 | 1.05E+03 | 1.10E-01 | 2.21E-04 | 6.08E-05 | 1.03E-09 | 6.23E-03 | 2.12E-04 | 3.94E-06 | 2.57E-03 | 41.8% |
| Pu-238 | 7.47E+02 | 7.85E-02 | 1.57E-04 | 3.82E-05 | 6.49E-10 | 3.92E-03 | 1.33E-04 | 2.48E-06 | 1.62E-03 | 26.3% |
| Pu-241 | 1.43E+04 | 1.50E+00 | 3.00E-03 | 1.59E-05 | 2.76E-10 | 1.63E-03 | 5.53E-05 | 1.03E-06 | 6.71E-04 | 10.9% |
| Pu-240 | 1.95E+02 | 2.05E-02 | 4.10E-05 | 1.11E-05 | 1.88E-10 | 1.14E-03 | 3.86E-05 | 7.17E-07 | 4.68E-04 | 7.6% |
| Cm-244 | 2.29E+02 | 2.40E-02 | 4.80E-05 | 6.87E-06 | 1.17E-10 | 7.04E-04 | 2.39E-05 | 4.45E-07 | 2.90E-04 | 4.7% |
| Pu-239 | 1.19E+02 | 1.25E-02 | 2.49E-05 | 6.73E-06 | 1.14E-10 | 6.90E-04 | 2.35E-05 | 4.36E-07 | 2.84E-04 | 4.6% |
| H-3 | 7.94E+01 | 7.94E+01 | 7.94E+01 | 2.65E-06 | 4.50E-11 | 2.72E-04 | 9.23E-06 | 1.72E-07 | 1.12E-04 | 1.8% |
| Sr-90 | 1.61E+04 | 1.69E+00 | 3.38E-03 | 2.34E-06 | 3.98E-11 | 2.40E-04 | 8.16E-06 | 1.52E-07 | 9.90E-05 | 1.6% |
| Kr-85 | 7.68E+02 | 7.68E+02 | 7.68E+02 | 4.34E-07 | 7.36E-12 | 4.45E-05 | 1.51E-06 | 2.81E-08 | 1.83E-05 | 0.3% |
| Am-243 | 5.41E+00 | 5.68E-04 | 1.14E-06 | 3.13E-07 | 5.31E-12 | 3.21E-05 | 1.09E-06 | 2.03E-08 | 1.32E-05 | 0.2% |
| Cs-137 | 2.27E+04 | 2.38E+00 | 4.77E-03 | 8.22E-08 | 1.40E-12 | 8.43E-06 | 2.86E-07 | 5.33E-09 | 3.47E-06 | 0.1% |
| | | | Total | 1.45E-04 | 2.47E-09 | 1.49E-02 | 5.07E-04 | 9.42E-06 | 6.14E-03 | 100% |
| | | | Total x 125 | 1.82E-02 | 3.09E-07 | 1.86E+00 | 6.33E-02 | 1.18E-03 | 7.68E-01 | |

Notes:
[1] - Based on values given in DOE-HDBK-3010-94
[2] - Based on nuclides expected to be present in spent nuclear fuel in storage in the FRS. Nuclides given here represent those that contribute greater than 0.1% of the TEDE.
[3] - Activity based on PWR fuel assemblies having 0.382 MTU per assembly.

## TABLE 9.2-3

## INADVERTENT CRITICALITY IN THE FRS

Assumptions:

| | |
|---|---|
| ARF x RF x DR x LPF (particulate) | Ref. Table 6-10, DOE-HDBK-3010-94 |
| ARF x RF x DR x LPF (fission gas) | 10% of Total Source Term, Table 6-7, DOE-HDBK-3010-94 |
| Release Height | 60 m |

| Receptor Location | | 640 m | 640 m | 640 m | 1050 m | 1050 m | 1700 m | |
|---|---|---|---|---|---|---|---|---|
| Stability Class, Wind Speed | | D, 4.5 m/s | F, 1 m/s | 95% | D, 4.5 m/s | F, 1 m/s | 95% | |
| Dispersion ($\chi/Q$) | | 1.59E-06 s/m$^3$ | 2.70E-11 s/m$^3$ | 1.63E-04 s/m$^3$ | 5.54E-06 s/m$^3$ | 1.03E-07 s/m$^3$ | 6.72E-05 s/m$^3$ | |
| Nuclide | Source Term (Ci) | On-Site Dose (rem) | On-Site Dose (rem) | On-Site Dose (rem) | Off-Site Dose (rem) | Off-Site Dose (rem) | Off-Site Dose (rem) | Percent Dose Contribution |
| Sr-90 | 3.42E+00 | 2.37E-03 | 4.03E-08 | 2.43E-01 | 8.26E-03 | 1.54E-04 | 1.00E-01 | 28.4% |
| Kr-89 | 4.10E+03 | 2.11E-03 | 3.58E-08 | 2.16E-01 | 7.34E-03 | 1.36E-04 | 8.90E-02 | 25.2% |
| Cs-134 | 4.43E+01 | 1.12E-03 | 1.90E-08 | 1.15E-01 | 3.90E-03 | 7.25E-05 | 4.73E-02 | 13.4% |
| Cs-137 | 3.16E+01 | 5.45E-04 | 9.26E-09 | 5.59E-02 | 1.90E-03 | 3.53E-05 | 2.30E-02 | 6.5% |
| Pu-241 | 7.25E-02 | 3.84E-04 | 6.52E-09 | 3.94E-02 | 1.34E-03 | 2.49E-05 | 1.62E-02 | 4.6% |
| Ru-106 | 1.53E+00 | 3.56E-04 | 6.05E-09 | 3.65E-02 | 1.24E-03 | 2.31E-05 | 1.50E-02 | 4.3% |
| Xe-138 | 1.10E+03 | 3.48E-04 | 5.91E-09 | 3.57E-02 | 1.21E-03 | 2.25E-05 | 1.47E-02 | 4.2% |
| Pu-238 | 1.29E-03 | 3.15E-04 | 5.35E-09 | 3.23E-02 | 1.10E-03 | 2.04E-05 | 1.33E-02 | 3.8% |
| Xe-137 | 4.90E+03 | 2.36E-04 | 4.01E-09 | 2.42E-02 | 8.23E-04 | 1.53E-05 | 9.98E-03 | 2.8% |
| Cm-244 | 1.38E-03 | 1.97E-04 | 3.34E-09 | 2.02E-02 | 6.85E-04 | 1.27E-05 | 8.31E-03 | 2.4% |
| Pu-240 | 3.12E-04 | 8.42E-05 | 1.43E-09 | 8.63E-03 | 2.93E-04 | 5.45E-06 | 3.56E-03 | 1.0% |
| Pu-239 | 1.87E-04 | 5.04E-05 | 8.55E-10 | 5.16E-03 | 1.76E-04 | 3.26E-06 | 2.13E-03 | 0.6% |
| H-3 | 1.22E+03 | 4.08E-05 | 6.92E-10 | 4.18E-03 | 1.42E-04 | 2.64E-06 | 1.72E-03 | 0.5% |
| Kr-88 | 6.60E+01 | 3.76E-05 | 6.38E-10 | 3.85E-03 | 1.31E-04 | 2.43E-06 | 1.59E-03 | 0.4% |
| I-134 | 4.80E+01 | 3.61E-05 | 6.14E-10 | 3.71E-03 | 1.26E-04 | 2.34E-06 | 1.53E-03 | 0.4% |
| Xe-135m | 3.30E+02 | 3.57E-05 | 6.07E-10 | 3.66E-03 | 1.25E-04 | 2.31E-06 | 1.51E-03 | 0.4% |
| Am-241 | 9.79E-05 | 2.70E-05 | 4.58E-10 | 2.76E-03 | 9.39E-05 | 1.75E-06 | 1.14E-03 | 0.3% |
| Kr-87 | 1.00E+02 | 2.25E-05 | 3.82E-10 | 2.31E-03 | 7.84E-05 | 1.46E-06 | 9.52E-04 | 0.3% |
| I-135 | 1.20E+01 | 1.20E-05 | 2.04E-10 | 1.23E-03 | 4.19E-05 | 7.79E-07 | 5.08E-04 | 0.1% |
| I-133 | 3.50E+00 | 1.05E-05 | 1.79E-10 | 1.08E-03 | 3.68E-05 | 6.83E-07 | 4.46E-04 | 0.1% |
| Co-60 | 3.43E-01 | 5.68E-06 | 9.65E-11 | 5.83E-04 | 1.98E-05 | 3.68E-07 | 2.40E-04 | 0.1% |
| | Total TEDE | 8.36E-03 | 1.42E-07 | 8.57E-01 | 2.91E-02 | 5.41E-04 | 3.53E-01 | 99.8% |

Notes:
[1] - Nuclides not contributing at least 0.1% to the TEDE not reported in table.
[2] - Source term activity based on PWR fuel having 0.382 MTU per assembly.