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Chapter 7

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Code Manual for MACCS2: Volume 1, User's Guide

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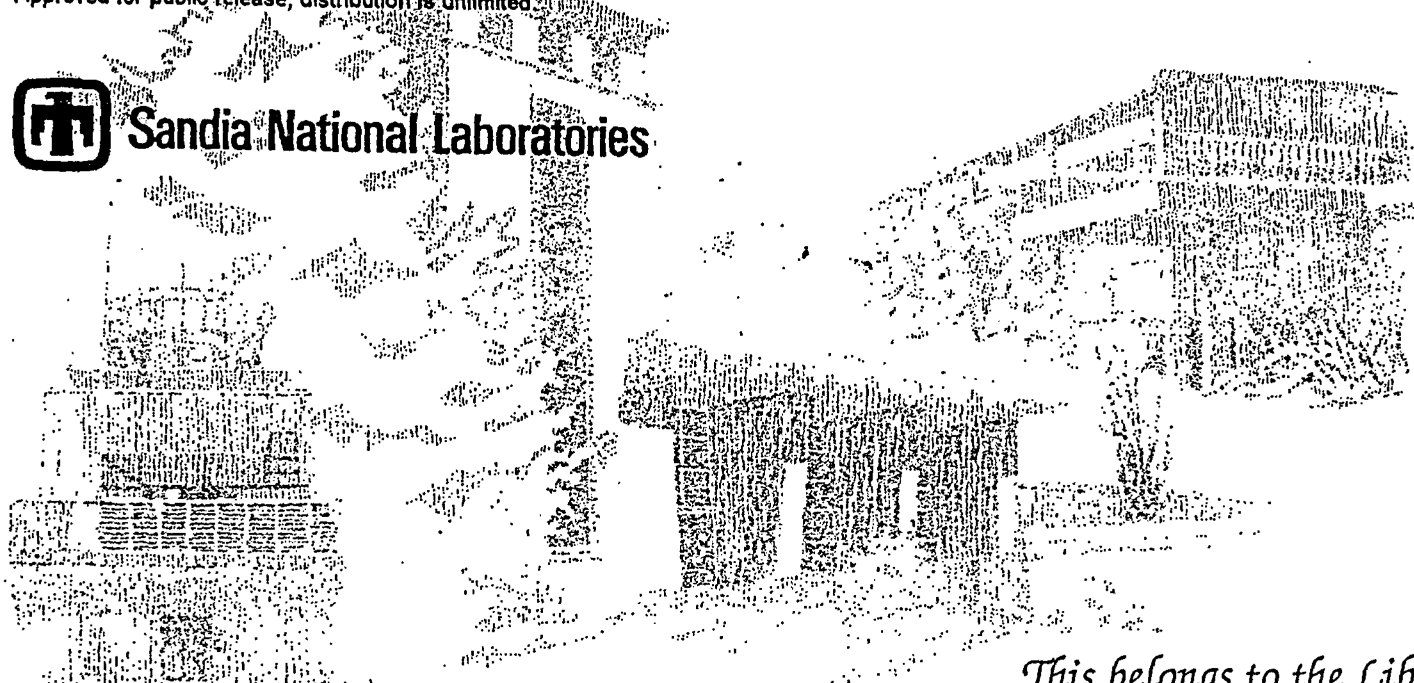

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Code Manual for MACCS2: Volume 1, User's Guide

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ABSTRACT

This report describes the use of the MACCS2 code. The document is primarily a user's guide, though some model description information is included. MACCS2 represents a major enhancement of its predecessor MACCS, the MELCOR Accident Consequence Code System. MACCS, distributed by government code centers since 1990, was developed to evaluate the impacts of severe accidents at nuclear power plants on the surrounding public. The principal phenomena considered are atmospheric transport and deposition under time-variant meteorology, short- and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs. No other U.S. code that is publicly available at present offers all these capabilities. MACCS2 was developed as a general-purpose tool applicable to diverse reactor and nonreactor facilities licensed by the Nuclear Regulatory Commission or operated by the Department of Energy or the Department of Defense. The MACCS2 package includes three primary enhancements: (1) a more flexible emergency-response model, (2) an expanded library of radionuclides, and (3) a semidynamic food-chain model. Other improvements are in the areas of phenomenological modeling and new output options. Initial installation of the code, written in FORTRAN 77, requires a 486 or higher IBM-compatible PC with 8 MB of RAM.

Melcor Accident Consequences Code System

EPRI 1999

ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT

VOLUME III
ALWR PASSIVE PLANT

CHAPTER 1
OVERALL REQUIREMENTS

Prepared For
Electric Power Research Institute
Palo Alto, California

Revision 8, Issued 3/99

VOLUME III, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5	**Methodology for Demonstrating Site Dose Criterion**	**Methodology for Demonstrating Dose Criterion**	8
	The demonstration that the site dose criterion is met shall use a physically-based source term release into an intact containment, as defined in Chapter 5, Section 2.4.1. The methodology for the PAG dose evaluation shall consist of the following;	The physically-based source term is based on release and removal phenomena from actual core damage sequences and should be reasonably bounding for source terms from the probabilistically significant sequences. The intact containment is based on ALWR containment performance requirements, which have been specified such that severe accident challenges to containment are effectively precluded or can be accommodated, thus providing integrity of the containment.	8
2.6.5.1	**Approach**	**Approach**	5
	A probabilistic dose (PD) method (e.g., CRAC2 or MACCS) shall be used.	A PD method is chosen for consistency with the basis for existing emergency planning and the fact that PD methods have provision for the particulate component of the source term (and thus are an appropriate method for calculating PAG comparison doses). The use of CRAC2, MACCS, or another similar code is consistent with current level 3 PRA evaluations and ALWR PRA Key Assumptions and Groundrules (KAG).	8
2.6.5.2	**Meteorological Database**	**Meteorological Database**	5
	The meteorological database shall be that provided in Annex B to Appendix A to Chapter 1 of the URD.	This meteorological database is that provided in the PRA KAG. It is an actual site meteorological database for which the RG 1.145 two-hour Exclusion Area Boundary X/Q is estimated to be greater than the X/Q for 80 to 90 percent of U.S. operating sites.	5

VOLUME III, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.3	**Direction-Dependent vs. Direction-Independent**	**Direction-Dependent vs. Direction-Independent**	5
	The dose calculation shall be direction-independent.	The calculations supporting existing emergency planning are direction-independent, i.e., the frequency of exceeding given dose levels is provided independent of direction. The NRC safety goals use a direction-independent approach as well. The use of a direction-independent approach is also consistent with the methods to be used in preparing the complementary cumulative distribution function (CCDF) for the exceedance frequency of off-site doses at the site boundary required by the PRA KAG.	5
2.6.5.4	**Statistical Measure of Dose to be Compared to 1 Rem PAGs**	**Statistical Measure of Dose to be Compared to 1 Rem PAGs**	8
	The dose to be compared to the 1 Rem PAG for ALWR emergency planning shall be the median dose.	Existing emergency planning on establishing that "most" core melt accidents would not exceed the PAG. There are two sources of variability in determining the meaning of "most" in the situation for existing emergency planning (i.e., NUREG 0396): the magnitude of the source term, and the meteorology. A similar approach is used here for ALWR emergency planning. Median dose (i.e., 50th percentile meteorology) together with the physically-based source term, which tends to bound the source term expected for nearly all core melt accidents in an ALWR, assures that the dose from most core melt accidents will not exceed 1 rem.	8

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Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.5	**Statistical Measure of Dose to be Compared to 5 Rem PAGs** The dose to be compared to the 5 PAG shall be the 90th percentile dose.	**Statistical Measure of Dose to be Compared to 5 Rem PAGs** More extreme (e.g., very stable atmospheric conditions, low wind speed) meteorology could cause higher doses for a given source term. While doses exceeding 1 rem would not be expected as noted above, a 5 rem limit has been specified for 90th percentile meteorology in order to address more extreme meteorological conditions. A 5 rem limit for such conditions is considered reasonable on several grounds. First, ICRP 63 recommends a dose limit for evacuation no lower than 50 mSv (i.e., 5 rem). Second, under stable, low wind speed conditions, the plume is concentrated (only about 100 feet wide at 0.5 mile) and is moving slowly, so the need for rapid evacuation would be quite limited. Finally, 5 rem is the upper end of the 1 to 5 rem range recommended by EPA and thus is a reasonable limit for emergency planning purposes under low probability weather conditions.	8

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.6	**Whole Body Dose vs. Effective Dose Equivalent**	**Whole Body Dose vs. Effective Dose Equivalent**	8
	The dose to be calculated is the sum of the effective dose equivalent (EDE) resulting from exposure to external sources (cloud shine and ground shine) and the committed effective dose equivalent (CEDE) from plume inhalation.	The May 1992 revision to Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (PAG Manual) calls for the use of EDE as the basis for determining off-site doses in relation to the 1 to 5 rem PAG. MACCS already employs this concept, as does the current 10CFR20. A separate thyroid dose limit is unnecessary since the EDE includes the organ weighted contribution from thyroid exposure. Not specifying a separate thyroid dose limit is also consistent with the recent NRC 10CFR50/100 rule change which specifies EDE as defined in Section 2.6.5.6. A separate ingestion exposure pathway requirement has not been specified since the ingestion exposure planning distance will be determined, using the May, 1992 PAG Manual guidelines, on a generic basis for all ALWRs. This will be accomplished by assuming that the 0.5 mile dose is equal to the PAG (i.e., the EDE limit for plume exposure); determining the maximum iodine contribution to this dose; and using this maximum iodine release as the basis for calculating the distance at which the ingestion dose equals the controlling ingestion pathway PAG (i.e., a projected infant thyroid dose from cows' milk of 1.5 rem on a preventative basis and 15 rem as a basis, for emergency contamination).	8

VOLUME III, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.7	**Inclusion of Organic Iodide in the PAG Calculation**	**Inclusion of Organic Iodide in the PAG Calculation**	8
	In calculating doses for comparison with the PAG values to justify ALWR emergency planning, the contribution from organic iodide can be neglected.	The I and HI are quite reactive and are likely to undergo natural deposition as rapidly (or more rapidly) than the particulate. Given that pH is controlled as specified in the Utility Requirements Document, the actual dose contribution from organic iodide is expected to be very small (a few percent of thyroid dose) and thus can be omitted from the dose calculation.	8
2.6.5.8	**Dose Commitment**	**Dose Commitment**	5
	A dose commitment of 50 years shall be used for TEDE from plume inhalation.	In the May 1992 revision of the PAG Manual, plume inhalation dose commitment is assumed to be the "lifetime". It is judged that a 50-year commitment is adequate on a generic basis to fulfill that requirement; it is also the duration used in the current 10CFR20.	8
		This differs from the PRA as specified in the KAG where the intent is to compare calculated doses to the 25 rem threshold for acute health effects (based on the current 25 rem whole body requirement in 10CFR100). It also differs from NUREG-0396 which uses one year commitment for inhalation.	8
2.6.5.9	**Radionuclides to be Included**	**Radionuclides to be Included**	5
	The radionuclides identified in Table II-2 of the CRAC2 User's Guide (NUREG/CR-2326) shall be the minimum list of radionuclides included in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning.	There are 54 radionuclides identified in this list. In MACCS there are six additional radionuclides: Sr-92, Y-92, Y-93, Ba-139, La-141, and La-142. These are not critical for the PAG comparison calculation; the impact of the Sr, Y, Ba and La isotopes already included in the CRAC2 list is much greater, given their relative quantities, half-lives and dose conversion factors; therefore, the CRAC2 list is acceptable.	5

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Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.10	**Dose Conversion Factors**	**Dose Conversion Factors**	5
	External dose conversion factors (plume and ground exposure) shall be based on Kocher, D.C., "*Dose Rate Conversion Factors for External Exposure to Photons, and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities,*" Health Phys., Volume 38, pp. 543-621 (1980). Inhalation dose conversion factors shall be based on Federal Guidance Report No. 11, "*Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion,*" Office of Radiation Programs, USEPA (1988).	Federal Guidance Report No. 11 is the document referenced by the May 1992 revision of the PAG Manual. However, in this guide, external dose conversion factors are provided only for noble gases. The external dose conversion factors used in MACCS for NUREG-1150 calculations are referenced in NUREG/CR-4551 to the specified Health Physics article. These are judged to be acceptable for the use described herein. The inhalation dose conversion factors provided in the guide are for a 50-year "lifetime" commitment, consistent with 2.6.5.8 above.	8
2.6.5.11	**Plume Modeling**	**Plume Modeling**	5
	The model used to treat dispersion in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning shall be a straightline Gaussian plume. Plume centerline doses shall be reported. The values of σy and σz that are used to characterize the Gaussian plume expansion shall be based on Pasquill-Gifford curves. If the analytical model used in the analysis employs a uniform approximation of the expansion in the crosswind (y) direction (e.g., CRAC2), the final result shall be increased by an appropriate factor to provide centerline doses. In the case of CRAC2 (which employs a 3-σy "top hat" approximation of the crosswind Gaussian distribution), the factor shall be 1.2.	The plume modeling in MACCS differs somewhat from that in CRAC2. The differences have been resolved as follows: • To demonstrate that the PAGs will not be exceeded within the exclusion area boundary (EAB) radius, the peak centerline value is the value that should be reported. To obtain this value, the CRAC2 results must be multiplied by a factor of 1.2. In addition, to compensate for the initially more disperse plume in CRAC2 (which results from setting the initial σy equal to building width/3 instead of building width/4.3), it is necessary to set the CRAC2 building width at the input level to 70% of its actual value.	8
	The initial σy shall be the building width divided by 4.3 if some other factor is used to determine the initial σy (e.g., a factor of 3 in CRAC2), and the building width specification shall be changed at the input level to compensate (e.g., the building width for CRAC2 shall be input as 70% of its actual value).		5

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.11	**Plume Modeling (Continued)**	**Plume Modeling (Continued)**	5

Requirement:

The correlation for dispersion in the vertical direction (z) shall be the form $\sigma_z = a_x b + c$ where x is the distance the plume has traveled. The values for a, b and c shall be the fixed values in CRAC2. In the event a simpler form has been employed for calculational ease (e.g., $\sigma_z = a_x b$ in MACCS), the coefficients shall be set to provide the same value of σz at a site boundary of 0.5 mile and at a low population zone (LPZ) radius of two miles as would be calculated using the fixed values for a, b and c in CRAC2. Those values are as follows:

Stability	a	b
A	2.47E-4	2.118
B	0.078	1.085
C	0.144	0.911
D	0.368	0.6764
E	0.2517	0.6720
F	0.184	0.6546

Rationale (Rev. 8):

- In CRAC2, the expansion in the z-direction (vertical) is controlled by an expression for σ_z as a function of plume travel, x. The expression has the form $\sigma_z = a_x b + c$ with the constants fixed in the coding. In MACCS, a different correlation which does not use an additive constant ("c" term) has been employed, but only for the purpose of convenience. For specific radial intervals of interest, values of a and b can be defined to give the same values of X/Q as CRAC2 at the two specific radial distances that define the interval. This is what has been done in this methodology specification. The 0.5-mile site boundary and 2-mile LPZ were chosen simply as typical radial distances.

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.11	**Plume Modeling (Continued)**	**Plume Modeling (Continued)**	5
	The time base for plume meander for long duration releases shall be the fixed value in CRAC2, three minutes.	• For long release times (greater than a few minutes), plume meander becomes an important factor in determining peak centerline doses. In CRAC2, the time base for plume meander was fixed at 3 minutes; in MACCS, it is a user input with 10 minutes having been used in NUREG-1150 and appearing in the standard problem input file. The data base supporting the modeling of plume meander includes averaging times (i.e., the time base) of approximately 3 to 10 minutes. Since the important parameter for plume meander is the ratio of release duration to the time base, and since the release duration being used in the PAG assessment is 10 hours, (per 2.6.5.14), duration to time base is better approximated by using the low end of the averaging range (i.e., the fixed CRAC2 value of 3 minutes) than the high end.	5
2.6.5.12	**Release Height and Energy of Release**	**Release Height and Energy of Release**	5
	The release height and energy of release assigned to the physically-based source term shall correspond to a cold, ground-level release for the purpose of calculating the dose.	Current severe accident analysis practice is to use release height and energy values that are consistent with the containment failure size/location or leak rate and associated thermodynamic conditions. However, for the ALWR physically-based source term, containment is intact, releases are not credited through a stack, and best estimate meteorology is used. Thus a cold, ground level release is appropriate.	5

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Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.13	**Duration of Exposure to Ground Contamination**	**Duration of Exposure to Ground Contamination**	5
	The duration of exposure to ground contamination shall be 24 hours from the start of release of fission products from the fuel.	The 24-hour period provides margin for ALWR accident detection, notification, and evacuation The 24-hour period is also consistent with the existing emergency planning basis.	8
2.6.5.14	**Duration of Release and Number of Plume Segments**	**Duration of Release and Number of Plume Segments**	5
	The release duration to be used in calculating doses for the ALWR physically-based source term shall be 10 hours if a single plume segment is used, or 24 hours if multiple plume segments are used.	The CRAC2 code has a limit on release duration of 10 hours and can employ only a single plume. The MACCS code will accept a release duration greater than 10 hours and can employ multiple plumes (i.e., different source terms in succession), - - this capability being most useful when the character of the release to the environment abruptly changes in the course of an accident. This is not the case for the ALWR physically-based source term, where the difference in dose between a 10-hour release duration and a 24-hour release duration is only a few percent.	8
2.6.5.15	**Shielding Factors**	**Shielding Factors**	5
	Shielding factors shall be 0.75 for plume exposure and 0.33 for exposure to ground contamination.	The values given are those from NUREG-0396, Section F, "no immediate protective actions" and are consistent with the "normal activity" requirement of the PRA KAG.	5

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Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.16	**Breathing Rate and Inhalation Protection Factors**	**Breathing Rate and Inhalation Protection Factors**	5
	The breathing rate shall be 3.3 x $10^{-4}m^3$/sec. For codes with provision for an inhalation protection factor, this value shall be set at 0.4. For codes without an inhalation protection factor, the breathing rate shall be reduced by a factor of 2.5.	The breathing rate identified in the May 1992 revision of the PAG Manual is the value specified. In the MACCS code, there is provision to reduce the inhalation dose by a factor to account for differences between the plume concentration and the concentration actually being breathed. NUREG/CR-4551 (one of the supporting documents for NUREG-1150) suggests an annual average value of 0.4 for normal activity (0.2 for active sheltering). The use of a "normal activity" inhalation protection factor is consistent with the requirements of the PRA KAG.	8
2.6.5.17	**Dry Deposition Velocity**	**Dry Deposition Velocity**	5
	The dry deposition velocity shall be 1.0 cm/sec for iodine and 0.1 cm/sec for other particulates.	These values are those of the May 1992 revision of the PAG Manual. Current severe accident analysis practice is to use values of 1.0 cm/sec (NUREG-0396/CRAC2) to 0.3 cm/sec (NUREG-1150/MACCS); the PRA KAG does not establish a requirement for dry deposition velocity.	8

NRC 2005

V-201

NUREG/CR-6525, Rev. 1

SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program

Sandia National Laboratories

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

NUREG/CR-6525, Rev. 1

SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program

Manuscript Completed: July 2003
Date Published: August 2003

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Pepared for
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code Y6594

ENVIRONMENTAL ASSESSMENT BY THE U.S. NUCLEAR REGULATORY COMMISSION RELATING TO THE CERTIFICATION OF THE AP1000 STANDARD PLANT DESIGN DOCKET NO. 52-006

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

RELATING TO THE CERTIFICATION OF THE

AP1000 STANDARD PLANT DESIGN

DOCKET NO. 52-006

The U.S. Nuclear Regulatory Commission (NRC) has issued a design certification for the Advanced Passive 1000 (AP1000) design in response to an application submitted on March 28, 2002, by Westinghouse Electric Company, LLC (hereinafter referred to as Westinghouse). A design certification is a rulemaking; the Commission has decided to adopt design certification rules as appendices to Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 52).

The NRC has performed an environmental assessment (EA) of the environmental impacts of the proposed new rule and has documented its findings of no significant impact in accordance with the requirements of 10 CFR 51.21 and the National Environmental Policy Act of 1969 (NEPA), as amended. This EA also addresses the severe accident mitigation design alternatives (SAMDAs), that the NRC has considered as part of this EA for the AP1000 design. This EA does not address the site-specific environmental impacts of constructing and operating a facility, which references the AP1000 design certification at a particular site; such impacts will be evaluated as part of any application or applications for the siting, construction, or operation of a facility.

As discussed in detail in Section 4.0 of this EA, the NRC determined that issuing this design certification does not constitute a major Federal action significantly affecting the quality of the human environment. The basis for this finding of no significant impact is that the design certification would not authorize the siting, construction, or operation of a facility of an AP1000 reactor design. Rather, the certification would merely codify the AP1000 design in a rule that could be referenced in a construction permit (CP), combined license (COL), or operating license (OL) application. Further, because the certification is just a rule, it does not involve any resources that have alternative uses. Therefore, the NRC has not prepared an environmental impact statement (EIS) in connection with this action.

The NRC also reviewed Westinghouse's evaluation of SAMDAs that generically apply to the AP1000 design. On that basis, the NRC found that the evaluation provides reasonable assurance that there are no additional SAMDAs beyond those currently incorporated into the AP1000 design which are cost-beneficial, whether considered at the time of the approval of the AP1000 design certification or in connection with the licensing of a future facility referencing the AP1000 design certification, where the plant referencing this appendix is located on a site whose site parameters are within those specified in Appendix 1B of the AP1000 design control document (DCD). These issues are considered resolved for the AP1000 design.

ENVIRONMENTAL ASSESSMENT

1.0 IDENTIFICATION OF THE PROPOSED ACTION

The proposed action would certify the AP1000 design under Appendix D to 10 CFR Part 52. The new rule would allow prospective licensees to reference the certified AP1000 design as part of a combined license (COL) application under 10 CFR Part 52 or may allow for a construction permit (CP) application under 10 CFR Part 50.

2.0 THE NEED FOR THE PROPOSED ACTION

The NRC has long sought the safety benefits of commercial nuclear power plant standardization and early final resolution of design issues. The NRC plans to achieve these benefits by certifying nuclear plant designs. Subpart B to 10 CFR Part 52 allows for certification in the form of rulemaking of an essentially complete nuclear plant design.

The proposed action would amend 10 CFR Part 52 to certify the AP1000 design. The amendment would allow prospective licensees to reference the certified AP1000 design as part of a COL application under 10 CFR Part 52 or may allow for a CP application under 10 CFR Part 50. Those portions of the AP1000 design included in the scope of the certification rulemaking would not be subject to further safety review or approval in a COL proceeding. In addition, the design certification rule would eliminate the need to consider SAMDAs for any future facilities that reference the certified AP1000 design.

3.0 THE ENVIRONMENTAL IMPACT OF THE PROPOSED ACTION

Issuing an amendment to 10 CFR Part 52 to certify the AP1000 standard plant design would not constitute a significant environmental impact. The amendment would merely codify the NRC's approval of the AP1000 design (refer to NUREG-1793). Furthermore, because the amendment is a rule, it involves no resources that have alternative uses.

As described in Section 4.0 of this EA, the NRC reviewed alternatives to the design certification rulemaking and alternative design features for preventing and mitigating severe accidents. NEPA requires consideration of alternatives to show that the design certification rule is the appropriate course of action and to ensure that the design referenced in the rulemaking does not exclude any cost-beneficial design changes related to the prevention and mitigation of severe accidents. The NRC concludes that, unlike the proposed design certification rule, the alternatives to certification do not provide for resolution of issues.

Design certification is in keeping with the Commission's intent to make future plants safer than the current generation of plants, to achieve early resolution of licensing issues, and to achieve the safety benefits of standardization (refer to the Advanced Reactor (51 FR 24643), Standardization (52 FR 348803), and Severe Accident Policy Statements (50 FR 32138), and to 10 CFR Part 52). Through its own independent analysis, the NRC also concludes that Westinghouse adequately considered an appropriate set of SAMDAs and that none were cost-beneficial. Although Westinghouse made no design changes as a result of reviewing the SAMDAs, Westinghouse had already incorporated certain features in the AP1000 design on the basis of the probabilistic risk assessment (PRA) results. Section 4.2 of this EA gives examples of these features. These design features relate to severe accident prevention and mitigation, but were not considered in the SAMDA evaluation because they were already part of the AP1000 design (refer to Section 19.1.6.2 of NUREG-1793, "AP1000 Design Improvement as a Result of Probabilistic Risk Assessment Studies").

Finally, the design certification rule by itself would not authorize the siting, construction, or operation of a nuclear power plant. The issuance of a CP, early site permit (ESP), COL, or OL which references the AP1000 design will require a prospective applicant to address the environmental impacts of construction and operation at a specific site. The NRC will then evaluate the environmental impacts and issue an EIS in accordance with 10 CFR Part 51. However, the SAMDA analysis has been completed as part of this EA and can be incorporated by reference into an EIS related to siting, construction, or operation of a nuclear plant that references the AP1000 design.

4.0 ALTERNATIVES TO THE PROPOSED ACTION

The NRC has identified two alternatives to certifying the AP1000 design. The first alternative would be to take no action to approve the design under Subpart B of 10 CFR Part 52. As with the proposed action, this alternative would not have a significant impact on the quality of the human environment because it would not authorize the siting, construction, or operation of a facility.

In the second alternative, the NRC would approve the design, but would not certify the AP1000 design in a rulemaking. The NRC issued a final design approval for AP1000 under Appendix O to 10 CFR Part 52 on September 13, 2004. Therefore, although the NRC has approved the design, the design would not have finality in proceedings under 10 CFR Part 50 or 10 CFR Part 52, Subpart C and could be modified. As a result, the design could require re-evaluation as part of each application to construct and operate a facility of an AP1000 design at a particular site. This alternative would provide for early internal NRC resolution of design issues to the extent that the design would remain unchanged at the facility application stage, but may not obtain all of the benefits of standardization nor permit overall finality for the resolved design issues.

The NRC sees no advantage in these alternatives compared to the design certification rulemaking proposed for the AP1000 design. Although neither the alternative nor the proposed action (design certification rulemaking) would significantly affect the quality of the human environment, the proposed action achieves the benefits of standardization, permits early resolution of design issues, and provides finality in licensing proceedings for the resolved design issues (including SAMDAs) that are within the scope of the design certification. Therefore, the NRC concludes that neither of the alternatives to rulemaking would achieve the objectives that the Commission intends by certifying the AP1000 design pursuant to 10 CFR Part 52, Subpart B.

4.1 Severe Accident Mitigation Design Alternatives

Consistent with the objectives of standardization and early resolution of design issues, the Commission decided to evaluate SAMDAs as part of the design certification for the AP1000 design. In a 1985 policy statement, the Commission defined the term "severe accident" as an event that is "beyond the substantial coverage of design-basis events," including events where there is substantial damage to the reactor core (whether or not there are serious offsite consequences). Design-basis events are events analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the DCD.

As part of its design certification application, Westinghouse performed a PRA for the AP1000 design to achieve the following objectives:

- Identify the dominant severe accident sequences and associated source terms for the design.
- Modify the design, on the basis of PRA insights, to prevent or mitigate and reduce the risk of severe accidents.
- Provide a basis for concluding that all reasonable steps have been taken to reduce the chances of occurrence, and mitigate the consequences, of severe accidents.

Westinghouse's PRA analysis is described in Chapter 19 of the AP1000 DCD.

In addition to considering alternatives to the rulemaking process discussed in Section 3.0, applicants for reactor design certification, COLs, or CPs must also consider alternative design features for severe accidents consistent with the requirements of 10 CFR

Part 50, and with a court ruling related to NEPA. These requirements can be summarized as follows:

10 CFR 52.79 and 10 CFR 50.34(f)(1)(I)[1] requires the applicant to perform a plant/site-specific PRA, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.

The U.S. Court of Appeals decision, in *Limerick Ecology Action v. NRC*, 869 F.2d 719 (3rd Cir. 1989), effectively requires the NRC to consider certain SAMDAs in the environmental impact review performed under Section 102(2)(c) of NEPA with respect to the licensing for operation of nuclear power plants.

Although these requirements are not directly related, they share a common purpose to consider alternatives to the proposed design, to evaluate whether potential alternative improvements in the plant design might increase safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed. It should be noted that the Commission is not required to consider alternatives to the design in this EA. However, as a matter of discretion, the Commission has determined that considering SAMDAs concomitant with the rulemaking is consistent with the intent of 10 CFR Part 52 for early resolution of issues, finality for resolved design issues, and achieving the benefits of standardization.

In its decision in *Limerick Ecology Action v. NRC*, the Court of Appeals for the Third Circuit expressed its opinion that it would likely be difficult to evaluate SAMDAs for NEPA purposes on a generic basis for all nuclear power plants then licensed by the NRC. However, the NRC has determined that generic evaluation of SAMDAs for the AP1000 standard design is both practical and warranted for two significant reasons. First, the design and construction of

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[1]Although 10 CFR 50.34(f)(1)(I) by its terms does not apply to new construction permits (CP), the Commission's policy is that a CP applicant will be required to comply with 50.34(f)(1)(I).

all plants referencing the certified AP1000 design will be governed by the rule certifying a single design. Second, the site parameters specified in the rule and the AP1000 DCD establish the consequences for a reasonable enveloping set of SAMDAs for the AP1000 design. The low residual risk of the AP1000 design and the limited potential for further risk reductions provides high confidence that additional cost-beneficial SAMDAs would not be found for sites within the site parameter envelope assumed for the AP1000 EA of SAMDAs. If the actual parameters for a particular site exceed those assumed in the rule and the DCD, then SAMDAs must be re-evaluated in the site-specific environmental report and the EIS. If the actual parameters for a postulated site are bounded by those assumed in the rule and the DCD, then the SAMDA analysis can be incorporated by reference in the site-specific EIS.

4.2 Potential SAMDAs Identified by Westinghouse

To identify candidate design alternatives, Westinghouse reviewed the design alternatives for other plants including the CE System 80+. Westinghouse also reviewed the results of the AP1000 PRA and design alternatives suggested by AP1000 design personnel.

Westinghouse eliminated the following SAMDAs from further consideration because they are already incorporated in the AP1000 design:

hydrogen ignition system

reactor cavity flooding system

reactor coolant pump seal cooling (AP1000 has canned motor pumps)

reactor coolant system (RCS) depressurization

external reactor vessel cooling

non-safety-grade containment sprays

Several risk-significant enhancements to the AP600 design have also been incorporated in the AP1000 design and were therefore not further considered. These modifications are summarized below and discussed further in DCD Tier 2, Section 1B.1.5, "Summary of Risk Significant Enhancement."

- a change in the normal position of the two containment motor-operated recirculation valves (in series with squib valves) from closed to open to improve the reliability of opening these flowpaths
- a change in the emergency operating procedures (EOPs) to call for in-containment refueling water storage tank (IRWST) draining earlier in an event to improve the probability of successful operator action
- a change in the design of the IRWST vents to preferentially direct hydrogen releases to the IRWST pipe vents, where diffusion flames will not adversely impact the containment
- incorporation of a low-boron core to reduce the potential contribution of anticipated transient without scram (ATWS) events to plant risk
- addition of a third passive containment cooling system (PCS) drainline with a motor-operated valve (MOV) that is diverse from the air-operated valves (AOVs) used in the other two drainlines, to improve PCS reliability
- specification that two of the four squib valves in the recirculation lines be low-pressure-type valves, and the remaining two squib valves be high-pressure-type valves to reduce the contribution to core damage frequency (CDF) from common-cause failures (CCFs) of recirculation squib valves

On the basis of the screening, Westinghouse retained 14 potential SAMDAs for further consideration. This set of SAMDAs is the same as that considered for the AP600 design. DCD Tier 2, Section 1B.1.3, "Selection and Description of SAMDAs," describes the 14 design improvements as follows:

(1) Upgrade the chemical and volume control system (CVCS) for small loss-of-coolant accidents (LOCAs): The CVCS is currently capable of maintaining the RCS inventory for LOCAs for effective break sizes up to 0.97 cm (3/8 in.) in diameter. A design alternative involving the upgrade of the CVCS for small LOCAs would increase the capability of the CVCS, enabling it to maintain RCS inventory during small- and intermediate-size LOCAs (up to an effective break size of 15.2 cm (6 in.) in diameter). Implementation of this design alternative would require installation of IRWST and containment recirculation connections to the CVCS, as well as the addition of a second line from the CVCS pumps to the RCS.

(2) Filtered vent: This design alternative would involve the installation of a filtered containment vent, including all associated piping and penetrations. This modification would provide a means to vent containment to prevent catastrophic overpressure failures and would also provide a filtering capability for source term release. The filtered vent would reduce the risk of late containment failures that might occur after failure of the PCS. Note, however, that even if the PCS fails, it is expected that air cooling will limit the containment pressure to less than the ultimate pressure capacity of the containment under most environmental conditions.

(3) Self-actuating containment isolation valves (CIVs): Self-actuation of CIVs could be used to increase the likelihood of successful containment isolation during a severe accident. This design alternative would involve the addition of a self-actuating valve or the enhancement of the existing CIVs on normally open containment penetrations (i.e., penetrations that provide normally open pathways to the environment during power and normal shutdown conditions). The design alternative would provide for self-actuation in the event that containment conditions are indicative of a severe accident. Closed systems inside and outside containment, such as the normal residual

heat removal system (RNS) and component cooling, would be excluded from this design alternative. The actuation of CIVs would be automatically initiated in the event that containment conditions are indicative of a severe accident.

(4) Passive containment sprays: This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 design (in lieu of the non-safety-related active containment spray capability currently incorporated in the AP1000 design). Installation of the safety-grade containment spray system could result in an increase in the following three risk benefits:

- scrubbing of fission products, primarily for containment isolation failure
- alternative means for flooding the reactor vessel (in-vessel retention)
- control of containment pressure if the PCS fails

(5) Active high-pressure safety injection (HPSI) system: A safety-related active HPSI system could be added that would be capable of preventing a core melt for all events except the large-break LOCA and ATWS. Note, however, that this design alternative is not consistent with the AP1000 design objectives. The AP1000 would change from a plant with passive systems to a plant with passive and active systems.

(6) Steam generator (SG) shell-side heat removal system: This design alternative would involve the installation of a passive safety-related heat removal system to the secondary side of the SGs. This enhancement would provide closed-loop secondary-system cooling by means of natural circulation and stored water cooling, thereby preventing the loss of the primary heat sink given the loss of startup feedwater (SFW) and the passive residual heat removal (RHR) heat exchanger (HX).

(7) Direct SG relief flow to the IRWST: To prevent fission product release from bypassing containment during a steam generator tube rupture (SGTR) event (or to reduce the amount released), flow from the SG safety and relief valves could be directed to the

IRWST. An alternative, lower cost option would be to redirect flow only from the first-stage safety valve to the IRWST.

(8) Increased SG pressure capability: As an alternative to design alternative (7) above, another method could be used to prevent fission product release from bypassing containment during an SGTR event (or to reduce the amount). This alternative method would involve an increase of the SG secondary-side pressure capability and safety valve pressure setpoint to a level high enough to not allow an SGTR to cause the secondary-system safety valve to open. Although detailed analyses have not been performed, it is estimated that the secondary-side design pressure would have to be increased by several hundred pounds per square inch (psi).

(9) Secondary containment filtered ventilation: This design alternative involves the installation of a passive charcoal and high-efficiency particulate air filter system for the middle- and lower-annulus region of the secondary concrete containment (below elevation 135'-3"). Drawing a partial vacuum on the middle annulus via an eductor with motive power from compressed gas tanks would operate the filter system. This design alternative would reduce particulate fission product release from any failed containment penetrations.

(10) Diverse IRWST injection valves: In the current design, a squib valve in series with a check valve (CV) isolates each of the four IRWST injection paths. To provide diversity, a modification could be made to allow a different vendor to provide the valves in two of the lines. Such diverse IRWST injection valves would reduce the likelihood of CCFs of the four IRWST injection paths.

(11) Diverse containment recirculation valves: In both the AP600 and AP1000 designs, two of the four recirculation lines have a squib valve in series with a CV, and the remaining two recirculation lines have a squib valve in series with an MOV. This SAMDA involves

changing the recirculation valve specification to enable two of the four lines to use diverse squib valves. To provide diversity, a modification could be made to allow a different vendor to provide the squib valves in two lines. Alternatively, in the AP1000 design, Westinghouse has specified that two of the four recirculation squib valves be designated as the low-pressure type and the remaining two squib valves as the high-pressure type. The diverse containment recirculation valves incorporated in the AP1000 design are responsive to the intent of this SAMDA and will reduce the frequency of core melt due to CCF of the four containment recirculation lines.

(12) Ex-vessel core catcher: This design alternative would inhibit core concrete interaction (CCI), even if the debris bed dries out. The enhancement would involve the design of a structure in the containment cavity or the use of a special concrete or coating. The current AP1000 design incorporates a wet cavity design in which ex-vessel cooling is used to keep core debris within the vessel. In cases where reactor vessel flooding has failed, the PRA assumes that containment failure occurs from an ex-vessel steam explosion or CCI.

(13) High-pressure containment design: A high-pressure containment design would prevent containment failures from severe accident phenomena such as steam explosions and hydrogen detonation. This proposed containment design would have a design pressure of approximately 2.17 mPa (300 psig) and would include a passive cooling feature similar to the one in the existing containment design. Although the high-pressure containment would not reduce the frequency or magnitude of releases from an unisolated containment, it would reduce the likelihood of containment failures.

(14) Increase reliability of diverse actuation system (DAS): The DAS is a non-safety system that can automatically trip the reactor and turbine and actuate certain engineered safety feature (ESF) equipment if the protection and safety monitoring system (PMS) is unable

to perform these functions. The DAS provides diverse monitoring of selected plant parameters to guide manual operation and to confirm reactor trip and ESF actuations. Increasing the reliability of the DAS involves adding a third instrumentation and control (I&C) cabinet and a third set of DAS instruments to allow the use of two-out-of-three logic instead of two-out-of-two logic.

Westinghouse considered an additional SAMDA that would involve relocating the entire normal residual heat removal system (RNS) and piping inside the containment pressure boundary. This would prevent containment bypass due to intersystem loss-of-coolant accidents (ISLOCAs) in the RNS. However, in the AP1000, the RNS has a higher design pressure than the systems in current pressurized-water reactors (PWRs), and an additional isolation valve is provided. As a result, ISLOCAs do not contribute significantly to the CDF in the AP1000 PRA. Accordingly, Westinghouse did not further investigate this change. The NRC has reviewed the Westinghouse analyses and agrees that further consideration of this change is not warranted because the change would provide virtually no risk reduction.

4.3 NRC Evaluation

The set of potential design improvements considered for the AP1000 is the same as those considered for the AP600. As part of the review for the AP600, the NRC reviewed the set of potential design improvements identified by Westinghouse and found it to be reasonably complete. The activity was accomplished by reviewing design alternatives associated with the following plants: Limerick, Comanche Peak, CE System 80+, Watts Bar, and the advanced boiling water reactor (ABWR). The NRC also reviewed accident management strategies described in (NUREG/CR-5474) and alternatives identified through the Containment Performance Improvement (CPI) Program (NUREG/CR-5567, -5575, -5630, and -5562). The results of this assessment are summarized in Appendix A to "Review of Severe Accident Mitigation Design Alternatives (SAMDAs) for the Westinghouse AP600 Design," Science and Engineering Associates, Inc., (SEA 97-2708-010-A;1, August 29, 1997). Given the similarity between the AP1000 and the AP600 design features and risk profile, the NRC considers this prior evaluation for the AP600 to be applicable to the AP1000 as well.

The NRC notes that the AP1000 design is less tolerant of equipment failures than the AP600 because the large LOCA success criterion for the AP1000 requires operation of two of two accumulators whereas only one of two accumulators is required for the AP600, and because the LOCA success criterion for the AP1000 requires operation of three of four automatic depressurization system (ADS) Stage 4 valves whereas only two of four ADS Stage 4 valves are required for the AP600. At the NRC's request, Westinghouse performed an evaluation of the two additional design alternatives:

(1) Larger accumulators: An increase in the size of the accumulators sufficient to change the large LOCA success criterion from two of two accumulators to one of two accumulators. Westinghouse estimates that the accumulator tanks would have to increase in size from 56.6 m^3 to 113.2 m^3 (2000 ft^3 to 4000 ft^3). This increase would likely require a change to the design of the direct vessel injection (DVI) piping subsystem and significant reanalysis of the DVI piping.

(2) Larger ADS Stage 4 valves: Increasing the size of the ADS Stage 4 (ADS-4) valves sufficient to change the LOCA success criterion from three of four valves to two of four valves. Westinghouse estimates that the valves would have to increase in size from 35.6 cm to 45.7 cm (14 in. to 18 in.) and that common fourth stage piping that connects to the hot leg would have to increase in size from 45.7 cm to 50.8 cm (18 in. to at least 20 in.). This increase would require a significant redesign of the squib valve and the ADS-4 piping, which in turn would impact the design of the reactor coolant loop piping.

Such a redesign would necessitate additional confirmatory testing to verify that the behavior of the passive safety systems was not adversely impacted.

For both of these alternatives, Westinghouse estimated that the redesign and reanalysis cost of the changes would be significantly greater than the benefits of completely eliminating all severe accident risk for the AP1000. Therefore, these design changes were not pursued further.

Although Westinghouse's analysis omitted several design alternatives, in most instances these design alternatives are either already included in the AP1000 design or bounded in terms of risk reduction by one or more of the design alternatives that were included in Westinghouse's analysis. In some other cases, design alternatives were pertinent only to boiling-water reactors (BWRs). The NRC's review did not reveal any obvious additional design alternatives that should have been considered by Westinghouse. Westinghouse considered some of the potential design alternatives identified in the above references as appropriate for accident management strategies, rather than as design alternatives. The NRC notes that the set of design improvements is not all inclusive in that additional, perhaps less expensive design improvements could be postulated. However, the benefits of any additional modifications would not likely exceed the costs of the modifications evaluated. Also, the costs of alternative improvements are not expected to be less than the costs of the least expensive improvements evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The discussions in DCD Tier 2, Appendix 1B, do not provide Westinghouse's basis or process for screening the many possible design alternatives to arrive at the final list of 14. Although the information provided does not demonstrate that the search for design alternatives was comprehensive, the NRC's review of the more than 120 candidate design alternatives considered for the AP600 did not identify any new alternatives more likely to be cost-beneficial than those included in the AP1000 design alternative evaluations. The NRC notes that Westinghouse has incorporated several risk significant enhancements within the AP1000 design, as discussed in Section 19.4.3.1 of NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," (AP1000 FSER), and has considered potential design changes to improve the AP1000 success criteria. On this basis, the NRC concludes that the set of potential design improvements evaluated by Westinghouse is acceptable.

4.4 Risk Reduction Potential of SAMDAs

4.4.1 Westinghouse Evaluation

In its evaluation, Westinghouse assumed that each design alternative would work perfectly to completely eliminate all severe accident risk from evaluated internal, external, and shutdown events. This assumption is conservative, since it maximizes the benefit of each design alternative. The design alternative benefits were estimated on the basis of the reduction of risk expressed in terms of whole body person-rem per year received by the total population within a 80.5-km (50-mile) radius of the AP1000 plant site, as discussed in Section 19.4.2 of the AP1000 FSER.

Westinghouse used the cost-benefit methodology of NUREG/BR-0184 to calculate the maximum attainable benefit of completely eliminating all risk for the AP1000. This methodology includes consideration of replacement power costs. The applicant estimated the present worth of eliminating all risk to be $21,000. Even if the AP1000 CDF and large release frequency (LRF) were a factor of 10 higher, this value would only increase to about $200,000.

4.4.2 NRC Evaluation

NRC reviewed Westinghouse's bases for estimating the risk reduction for the various SAMDAs, and concluded that Westinghouse used bounding and conservative assumptions as the bases for the risk reduction estimates for each design alternative.

Westinghouse's risk reduction estimates are based on point-estimate (mean) values, and do not consider uncertainties in CDF or offsite consequences. Although this is consistent with the approach taken in previous design alternative evaluations, further consideration of these factors could lead to significantly higher risk reduction values, given the extremely small CDF and risk estimates in the baseline PRA. In assessing the risk reduction potential of design improvements for the AP1000, the NRC has based its evaluation on the applicant's risk reduction estimates for the various design alternatives, in conjunction with an assessment of the potential impact of uncertainties on the results. This assessment is discussed further in Section 19.4.6 of the AP1000 FSER and in Section 4.6 of this EA.

4.5 Cost Impacts of Candidate SAMDAs

4.5.1 Westinghouse Evaluation

DCD Tier 2, Section 1B.1.8, "Evaluation of Potential Improvements," discusses capital cost estimates for the design alternatives evaluated by Westinghouse for the AP1000. DCD Tier 2, Table 1B-5, presents the results of the cost evaluations. The cost evaluations did not account for the costs of design engineering, testing, and maintenance for each design alternative. Including these costs would increase the overall costs and decrease the benefits of each alternative. Thus, the Westinghouse approach is conservative.

4.5.2 NRC Evaluation

As mentioned previously, the set of SAMDAs considered for the AP1000 is the same as the set considered for the AP600. As part of the AP600 review, the NRC compared the capital costs for the AP600 design alternatives with those evaluated for the ABWR and CE System 80+ designs. The purpose of this comparison was to determine the reasonableness of the cost estimates presented by the applicant. The design alternatives among the reactor designs, did not exactly match, so only rough comparisons were possible. Based on these comparisons, the NRC concluded that the cost estimates for the AP600 design alternatives are in reasonable agreement with the costs for roughly similar design alternatives evaluated for other plants. Given the similarity between the AP1000 and the AP600 design features and risk profile, the NRC considers this prior evaluation for the AP600 to be applicable to the AP1000 as well. This is reasonable, considering uncertainties in the cost estimates, and the level of precision necessary, given the greater uncertainty inherent on the benefit side with which these costs were compared.

4.6 Cost-Benefit Comparison

4.6.1 Westinghouse Evaluation

After considering the risk reduction potential and cost impact of the various SAMDAs, Westinghouse did a cost-benefit comparison to determine whether any of the potential severe accident design features would be justified. To do so, Westinghouse evaluated the benefits of each design alternative in terms of potential risk reduction, which was defined as the reduction in whole body person-rem per year received by the total population within a 80.5-km (50-mile) radius of the AP1000 plant site. Westinghouse used the cost-benefit methodology of NUREG/BR-0184 to calculate the maximum attainable benefit of completely eliminating all risk for the AP1000. This methodology includes consideration of replacement power costs. Westinghouse estimated the present worth of eliminating all risk to be $21,000. This value is an upper bound because in practice no design alternative, if implemented, would reduce the plant CDF to zero. Westinghouse also provided additional sensitivity analyses of the impacts of the following:

- a 3-percent discount rate rather than the 7-percent discount rate assumed in the base case
- a factor of 10 increase in the population dose used in the base case
- a more realistic reduction in CDF (i.e., each SAMDA reduces CDF by 50 percent rather than 100 percent, as assumed in the base case)
- a factor of 2 increase in the base case CDF
- a factor of 10 increase in the maximum attainable benefit

DCD Tier 2, Table 1B-4, summarizes the results for these cases. With the exception of the last sensitivity case, the calculated maximum attainable benefit was no more than $43,000. Even when the AP1000 CDF and LRF were increased by a factor of 10, the maximum attainable benefit of eliminating all risk for the AP1000 would only increased to about $200,000.

The applicant found that none of the 14 design alternatives and neither of the two additional alternatives related to the PRA success criteria would be cost beneficial. Only one alternative has an implementation cost close to $21,000, namely, SAMDA 3, self-actuating CIVs, which has an estimated cost of $33,000. All of the remaining alternatives have estimated implementation costs at least a factor of 20 greater than the maximum attainable benefit of

$21,000. On this basis, the applicant concluded that only SAMDA 3 warranted further evaluation.

SAMDA 3 consists of improved containment isolation provisions on all normally open containment penetrations. The design alternative would involve either adding a self-actuating valve or enhancing the existing inside CIV to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Westinghouse noted that even if this SAMDA completely eliminated all releases associated with containment isolation failures (i.e., release category containment isolation (CI)) and reduced the CDF to zero, the benefit of the SAMDA would be on the order of $1000. More realistically, the CDF would not be impacted, and elimination of all containment isolation failures would only have a benefit on the order of $100. Thus, even the lowest cost SAMDA would not be cost beneficial.

On the basis of the cost-benefit comparison, the applicant concluded that no additional modifications to the AP1000 design were warranted.

4.6.2 NRC Evaluation

The applicant's estimates of risk do not account for uncertainties either in the CDF or in the offsite radiation exposures resulting from a core damage event. The uncertainties in both of these key elements are fairly large because key safety features of the AP1000 design are unique and their reliability has been evaluated through analysis and testing programs rather than operating experience. In addition, the estimates of CDF and offsite exposures do not account for the added risk from earthquakes.

As part of the AP600 review, the NRC did detailed analyses to assess design alternative benefits, taking into account the uncertainties in estimated CDF, offsite releases of radioactive materials from a severe accident, and the effects of external events. Given the similarities between the AP1000 and AP600 design features and risk profiles and the sets of SAMDAs relevant to each design, the NRC considers this prior evaluation for the AP600, summarized below, to be applicable to the AP1000 as well.

The staff estimated the maximum benefits that could be achieved with the AP600 design alternatives, assuming that a design alternative can either completely eliminate all core damage events or completely eliminate offsite releases of radioactive materials in the event of a severe accident. The estimates of benefits were calculated using the NRC-developed FORECAST code (NUREG/CR-5595, Revision 1, "FORECAST: Regulatory Effects Cost Analysis Software Manual, Version 4.1," Science and Engineering Associates, Inc., July 1996). FORECAST allows the use of uncertainty ranges for all key parameters and provides a means for combining uncertainties in these parameters. For the purposes of estimating the maximum potential benefit from the AP600 design alternatives, the staff assumed that external events and accident sequences not yet accounted for in the PRA increased the reference CDF by two orders of magnitude (i.e., a factor of 100).

The results of the analysis indicated that design alternatives which prevent accidents (i.e., reduce the accident frequency to zero) are much more cost effective than design alternatives which reduce or eliminate offsite releases, but which have no effect on accident frequency. This is because of the fairly large benefits of averting onsite cleanup and decontamination costs and avoiding replacement energy costs. Neither of these costs are assumed to be impacted by design alternatives which do not reduce accident frequency. The staff divided the design alternatives into two groups: those that impact the CDF and those that impact containment performance, but not CDF. Benefits were estimated by taking the fractional reduction in risk for each design alternative (compared to the AP600 baseline risk as defined by the applicant) and applying that fraction to the mean benefits.

Design alternatives that were within a decade of meeting a benefit-cost criterion of \$5000/person-rem were subjected to further probabilistic and deterministic considerations. None of the design alternatives had a cost-benefit ratio of less than \$5000/person-rem. The only design alternatives which came within a decade of the \$5000/person-rem criterion were SAMDA 10, diverse IRWST injection valves, and SAMDA 3, self-actuating CIVs. The NRC concludes, on the basis of further probabilistic and deterministic evaluations, that these design alternatives are not cost beneficial and need not be further pursued.

Given the similarities between the AP1000 and the AP600 design features and risk profiles and the sets of SAMDAs relevant to each design, the NRC considers the results of this prior evaluation for the AP600 to be applicable to the AP1000 as well. Accordingly, the NRC further evaluated these two SAMDAs for the AP1000, as discussed below.

4.7 Further Considerations

4.7.1 Self-Actuating Containment Isolation Valves

This design alternative would reduce the likelihood of containment isolation failure by adding self-actuating valves or enhancing the existing CIVs for automatic closure when containment conditions indicate a severe accident has occurred. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions. For example, a fusible link would melt in response to elevated ambient temperatures, venting the air operator of a fail-closed valve, thus providing the self-actuating function. This design alternative is estimated to impact releases from containment by less than 10 percent.

This improvement to the containment isolation capability would appear to be effective in reducing offsite releases for accidents involving external and internal events. The addition of this design alternative would impose minor operational disadvantages to the plant because the operations and maintenance staff would require some additional training. These automatic features would also require periodic testing to assure that they were functioning properly.

The most important question regarding this design alternative is whether it can be implemented for a cost of only $33,000. The cost estimate appears not to include the first-time engineering and qualification testing that would be required to demonstrate that the valve would perform its intended function in a timely and reliable manner. The costs of periodic testing and maintenance appear not to have been included. The NRC believes that the actual costs of this design alternative would be substantially higher than the applicant's estimate (by a factor of 10 or more) when all related costs are realistically considered. On the basis of the unfavorable cost-benefit ratio and the expectation that actual costs would be even higher than the applicant estimated, the NRC concludes that this design alternative is not cost beneficial and need not be further evaluated.

4.7.2 Diverse IRWST Injection Valves

In the current AP1000 design, a squib valve in series with a CV isolates each of four IRWST injection paths. This design alternative would reduce the likelihood of CCFs of IRWST injection to the reactor by utilizing diverse valves in two of the four lines. The complete elimination of the CCFs of IRWST injection squib valves would lead to a moderate (up to 10 percent) reduction of the at-power internal events CDF. In the absence of a comprehensive external events PRA for the AP1000 plant, it is difficult to estimate the effectiveness of this design alternative in reducing the risk from external events such as seismic events. However, it appears likely that failure to inject coolant to the reactor would remain a contributor to the CDF from external events, in which case diversity in the IRWST injection valves should help to reduce the risk from both external and internal events.

Alternate vendors are available for the CVs. However, it is questionable if CVs of different vendors would be sufficiently varied to be considered diverse unless the type of CV was changed from the current swing-disk check valve type to another type. The swing-disk type is preferred for this application and other types are considered less reliable.

Adding diversity to the injection line squib valves would require additional spares at the plant and some additional training for plant operations and maintenance staff, but would not appear to add significantly to the operational aspects of the AP1000. However, a greater issue concerns the availability and costs of acquiring diverse valves from a second vendor. Squib valves are specialized valve designs for which there are few vendors. The applicant claimed that a vendor might not be willing to design, qualify, and build a reasonable squib valve design for this application, considering that the vendor would only supply two valves per plant. The cost estimate for this design alternative assumes that a second squib valve vendor exists and that the vendor only provides the two diverse IRWST squib valves per plant. The cost estimate does not include the additional first-time engineering and qualification testing costs that will be incurred by the second vendor. The applicant estimated that those costs could be more than $1 million dollars. As a result, the applicant concluded that this design alternative would not be practicable because of the uncertainty in the availability of a second squib valve design/vendor and the uncertainty about the reliability of another type of CV. The NRC considers the rationale set forth by the applicant regarding the potential reductions in reliability and high costs associated with obtaining diverse valves to be reasonable. On the bases of these arguments, the NRC concludes that this design alternative need not be further pursued.

4.8 Conclusions on SAMDAs

As discussed in Section 19.1 of the AP1000 FSER, Westinghouse used the PRA results extensively to arrive at the final AP1000 design. As a result, the estimated CDF and risk calculated for the AP1000 design are very low, both relative to existing operating plants and in absolute terms. Moreover, the low CDF and risk for the AP1000 plant reflect Westinghouse's efforts to systematically minimize the effect of initiators/sequences that have been important contributors to CDF in previous PWR PRAs. This minimization has been done largely through the incorporation of a number of design improvements. Section 19.1 of the AP1000 FSER discusses these improvements and the additional AP1000 design features which contribute to low CDF and risk for the AP1000.

Because the AP1000 design already has numerous plant features designed to reduce CDF and risk, the benefits and risk reduction potential of additional plant improvements is significantly reduced. This reduction is true for both internally and externally initiated events. Moreover, with the features already incorporated in the AP1000 design, the ability to estimate CDF and risk approaches the limitations of probabilistic techniques. Specifically, when CDFs are estimated to be on the order of 1 in 1,000,000 years, it is possible that the areas of the PRA where modeling is least complete, or supporting data are sparse or even nonexistent, may actually be the more important contributors to risk. Areas not modeled or incompletely modeled include human reliability, sabotage, rare initiating events, construction and design errors, and systems interactions. Although improvements in the modeling of these areas may introduce additional contributors to CDF and risk, the NRC does not expect that additional contributions would change the conclusions in absolute terms.

The NRC concludes that none of the potential design modifications evaluated are justified on the basis of cost-benefit considerations. The NRC further concludes that it is unlikely that any other design changes would be justified in the future on the basis of person-rem exposure because the estimated CDFs are very low on an absolute scale.

5.0 ALTERNATIVE USE OF RESOURCES

No resources, such as land, water, or physical materials, will be affected by the promulgation of this proposed rule. This proposed rule would codify the AP1000 design in the *Code of Federal Regulations* but would not authorize the siting, construction, or operation of any nuclear power plant.

6.0 STATES CONSULTED AND SOURCES USED

The NRC sent a copy of the proposed rule and draft EA to the State Liaison Officers and specifically requested their comments on the EA. In addition, the draft EA was issued for public comment; comments and responses are discussed in Section 7.

The Commission has determined under the NEPA of 1969, as amended, and the NRC's regulations in 10 CFR Part 51, Subpart A, that this rule is not a major Federal action significantly affecting the quality of the human environment. Therefore, the NRC has determined that preparation of an environmental impact statement for this rulemaking is not required. The basis for this determination, as documented in this EA, is that the amendment to 10 CFR Part 52 would not authorize the siting, construction, or operation of a facility referencing the AP1000 design; it would only codify the AP1000 design in a rule. Therefore, the NRC staff did not issue the EA for comment specifically by Federal, other State, and local agencies. The NRC's finding of no significant environmental impact was published in the *Federal Register* on April 18, 2005 (70 FR 20062), with the proposed design certification rule and draft EA for the

AP1000 design. The NRC will evaluate the environmental impacts and issue an EIS, as appropriate, in accordance with NEPA as part of any application(s) for the siting, construction, or operation of a facility that would reference the AP1000 design..

7.0 PUBLIC COMMENTS AND NRC RESPONSES

On April 18, 2005 (70 FR 20062), the Commission issued the draft EA for public comment. The comment period expired on July 5, 2005. The comments are summarized below and responses are provided; the comments did not result in a change in the technical analyses, findings, or conclusions in the EA.

Comment summary. Three severe accident mitigation design alternatives (SAMDAs) were inappropriately dismissed in the EA on the basis that they do not affect the likelihood of an accident. These SAMDAs involve filtered containment vents and self-actuating containment isolation valves.

Response. The NRC disagrees that these three SAMDAs were inappropriately dismissed. The noted SAMDAs were assessed in terms of their respective benefits and implementation costs, and dismissed on the basis that they would not be cost-beneficial. In assessing benefits, SAMDAs were divided into two groups—those that impact core damage frequency (CDF), and those that impact containment performance but not CDF (including the SAMDAs in question). Although containment-related SAMDAs do not offer any benefits associated with reducing CDF (such as averted replacement power costs), the applicant conservatively assumed that all SAMDAs would completely eliminate all severe accident risk. More realistically, the CDF would not be impacted and the benefits would be much lower. Accordingly, these SAMDAs would not be cost-beneficial.

Comment summary. One SAMDA was inappropriately dismissed in the EA on the basis that it is not consistent with the AP1000 design objective of relying on passive systems. This SAMDA involves an active high-pressure safety injection system that would be capable of preventing a core melt for all but two types of events.

Response. The NRC disagrees that the SAMDA was inappropriately dismissed. Although the noted SAMDA was screened out on the basis that it is inconsistent with AP1000 design objectives, it would also have been eliminated on cost-benefit considerations. Specifically, even if this SAMDA were to eliminate all severe accident risk, the estimated costs of the SAMDA (at least $1 million, given the significant hardware and ongoing maintenance costs) would exceed the estimated benefits by several orders of magnitude.

Comment summary. The EA contains no assessment of the impact of an accidental or deliberate external rupture of the AP1000's unreinforced containment structure.

Response. For the reasons the Commission stated in detail in *Private Fuel Storage* (CLI-02-25, 12/18/2002), the NRC has no obligation under the National Environmental Policy Act (NEPA) to consider intentional malevolent acts, such as those directed against the United States on September 11, 2001, in conjunction with a licensing action. In short, the Commission recognizes that it cannot rule out the possibility of a terrorist threat to nuclear facilities, but finds that the possibility of a terrorist attack is speculative and simply too far removed from the natural or expected consequences of agency action to require a study under NEPA. As a practical matter, attempts to evaluate that threat even in qualitative terms are likely to be meaningless and consequently of no use in the agency's decision making. Moreover, although one of the purposes of NEPA is to inform the public of the environmental impacts of a regulatory action, the results of any attempted analysis of terrorism could not be made available to the public, for reasons associated with safeguards and physical security.

The Commission is devoting substantial time and agency resources to combating the potential for terrorism involving nuclear facilities and materials. In response to the September 11 attacks, the NRC staff is conducting a comprehensive review of its security and safeguards measures, and have instituted interim upgrades in security requirements for its licensees. The Commission is also working with numerous other government agencies to meet and minimize the threat of terrorism. Thus, although the Commission declines to consider terrorism in the context of NEPA, it is devoting significant attention to terrorism-related matters.

Comment summary. How can anyone do an "Environmental Assessment" or an FSER on a plant design that exists only on paper and has never been constructed completely to scale and operated anywhere in the world?

Response. The logical outgrowth of this argument is that no plant of new design could ever be built; the argument is circular. The purpose of an FSER and EA is to assess a nuclear plant design before it is constructed. The FSER is based on an evaluation of design information and the safety analyses of postulated accidents for that particular plant design. The SAMDA portion of the EA considers alternatives to the plant design that was evaluated in the FSER. The NRC's FSER and EA for the AP1000 standard plant design were used as the basis for this rulemaking.

Comment summary. The applicant's estimates of risk do not account for uncertainties in core damage frequency or in offsite radiation exposures resulting from a core damage event.

Response. The NRC disagrees with this comment. Although the NRC acknowledges that uncertainties are large and that several areas are incompletely modeled, as stated in the EA, even if the CDF and large release frequency were a factor of 10 higher, none of the SAMDAs would be cost-beneficial.

Comment summary. The Department of Energy (DOE) is going to subsidize "first of a kind" engineering costs for the first plants constructed of each of the new NRC-approved designs. Therefore, the applicant is not going to have to bear all costs considered in the analysis.

Response. The cost evaluations do not include the costs of design engineering or testing and maintenance for each design alternative. Including all or a portion of these costs would increase the overall implementation costs and decrease the cost-effectiveness of each SAMDA. Moreover, the possibility that DOE may pay for the "first of a kind" engineering costs for the first plants is not relevant, since that only addresses who is going to pay for such costs; the SAMDA analysis focuses on the overall cost to society.

Comment summary. There seems to be no inclusion in the cost-benefit analysis of the "benefit" to the applicant of a plant which has little or no severe accident risk. Westinghouse stands to gain significantly if the AP1000 is as safe as the AP600 is supposed to be.

Response. The comment appears to be based on the incorrect assumption that the SAMDA analysis and/or Regulatory Analysis should include benefits to an applicant utilizing the AP1000 design. The low level of risk estimated for the AP1000 design may be a benefit to the applicant with regard to marketability and public acceptance of the design. However, this is not a recognized or readily quantifiable attribute in the NRC methodology for value-impact analysis (NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook") and there is no precedent for its inclusion in regulatory analyses. Accordingly, this factor has not been included in the SAMDA evaluation.

Comment summary. The cost-benefit methodology overstates the costs and understates the benefits by including replacement power costs as part of the SAMDA implementation cost rather than as a benefit.

Response. The comment reflects a misunderstanding of how replacement power costs were treated in the assessment. Replacement power costs (more correctly, "averted replacement power costs") were included as a benefit for the various SAMDAs, and were not assumed to contribute to the SAMDA implementation costs.

Comment summary. The comment questions how one can estimate populations that are totally hypothetical, and why the entire population within a 50-mile radius of the plant is used in the analysis. The comment implies that use of the entire population would have the effect of diluting (reducing) the hypothetical exposure from an accident.

Response. Offsite consequences for the AP1000 design were evaluated using reference site information developed by the Electric Power Research Institute (EPRI) to represent potential sites where an AP1000 plant might be built. The reference site data was developed to represent or bound the consequences at approximately 80 percent of the reactor sites in the United States (see Section 19.4.2 of the AP1000 FSER). Exposure and offsite property impacts were estimated over a 50-mile radius from the plant site as prescribed in NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." The population dose estimates represent the cumulative dose received by the entire population within the 50-mile radius. Consideration of the entire population increases rather than dilutes the hypothetical exposure from an accident.

Comment summary. The NRC accepts the applicant's assessment when the estimated implementation costs are higher than the estimated benefits, yet rejects the applicant's cost estimates for SAMDAs whose implementation costs are within the range of the estimated benefits. One of the SAMDAs handled in this manner was self-actuating containment isolation valves.

Response. The NRC disagrees with the comment. The methodology for evaluating potential SAMDAs involves a multi-step screening process. SAMDAs whose implementation costs clearly exceed the conservatively-estimated benefits are screened from further consideration. Those SAMDAs whose implementation costs are within range of the estimated benefits are further assessed using more realistic assumptions regarding implementation costs and/or benefits. The SAMDA assessment for self-actuating containment isolation valves is an example of a SAMDA that survived the initial screening, but was subsequently judged not cost-beneficial under more realistic assumptions.

Comment summary. The SAMDA cost-benefit analysis is based on construction of a single unit, even though this design, once certified could be referenced for many plants. Thus, the costs of any re-engineering and re-analysis involved in the incorporation of any of the SAMDAs would effectively be spread over many plants.

Response. The staff agrees that the costs of any re-engineering and re-analyzing can be spread over many plants. However, this would not affect the measures of the SAMDA analysis because the applicant's cost estimates did not account for the costs of design engineering. Thus, most of the SAMDA implementation cost (e.g., the cost of installed hardware) would still be incurred at each unit regardless of whether additional units are constructed. In addition, even if all SAMDA implementation costs were assumed to be reduced by a factor of 10, to represent spreading all costs over 10 new units, none of the potential SAMDAs would become cost-beneficial when SAMDA benefits and implementation costs are estimated based on realistic assumptions.

Comment summary. The comment questions how cost considerations are allowed to influence the safety review and design certification process.

Response. The NRC disagrees that cost considerations have influenced the safety review. It is important to recognize the difference between the safety evaluation and the EA. The review of the AP1000 design with regard to the overall level of safety and its compliance with NRC's regulations is described in the AP1000 FSER. Costs are not an ordinary consideration in the NRC's safety evaluation, i.e., the design is required to meet all regulations regardless of cost unless an exemption is requested and costs are defined as a legitimate factor to be considered under one or more of the criteria in 10 CFR 50.12. In contrast, the scope and focus of the SAMDA review within the EA is on potential means by which plant risk can be further reduced. Costs are a legitimate consideration in this assessment, since the objective is to identify significant and practical improvements in plant design that do not impact excessively on the plant cost.

8.0 <u>FINDING OF NO SIGNIFICANT IMPACT</u>:

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has decided not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the design certification rule and the documents referenced in the statement of consideration for the final rule. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and

Management System (ADAMS) Public Electronic Reading Room on the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents in ADAMS should contact the NRC PDR reference staff at 1-800-397-4209 or 301-415-4737 or send an e-mail to pdr@nrc.gov.

Westinghouse 2004

AP1000

Probabilistic Risk Assessment, Revision 8

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The design, engineering, and other information contained in this document have been prepared by or on behalf of the Westinghouse Electric Company LLC in connection with its application to the United States Nuclear Regulatory Commission (NRC) for design certification of the AP1000 passive nuclear plant design pursuant to Title 10, Code of Federal Regulations Part 52. No use of or right to copy any of this information, other than by the NRC and its contractors in support of Westinghouse's application, is authorized.

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APPENDIX 1B

SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

1B.1 AP1000 SAMDA Evaluation

1B.1.1 Introduction

This response provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for the Westinghouse AP1000 design. This evaluation is performed to evaluate whether or not the safety benefit of the SAMDA outweighs the costs of incorporating the SAMDA in the plant, and is conducted in accordance with applicable regulatory requirements as identified below.

The National Environmental Policy Act (NEPA), Section 102.(C)(iii) requires, in part, that:

> ... all agencies of the Federal Government shall ... (C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on ... (iii) alternatives to the proposed action.

The 10 CFR 52.47(a)(ii) requires an applicant for design certification to demonstrate:

> ... compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) ...

A relevant requirement of 10 CFR 50.34(f) contained in subparagraph (1)(i) requires the performance of:

> ... a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant ...

In SECY-91-229, the U.S. Nuclear Regulatory Commission (NRC) staff recommends that SAMDAs be addressed for certified designs in a single rulemaking process that would address both the 10 CFR 50.34 (f) and NEPA considerations in the 10 CFR Part 52 design certification rulemaking. SECY-91-229 further recommends that applicants for design certification assess SAMDAs and the applicable decision rationale as to why they will or will not benefit the safety of their designs. The Commission approved the staff recommendations in a memorandum dated October 25, 1991 (Reference 1).

1B.1.2 Summary

Note that the AP1000 is similar to the AP600, which has received Design Certification. The evaluation for AP1000 uses the conclusions of the AP600 SAMDA investigation as described below. An evaluation of candidate modifications to the AP600 design was conducted to evaluate the potential for such modifications to provide significant and practical improvements in the

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7.3

(NRC 2005) U.S. Nuclear Regulatory Commission, *Environmental Assessment by the U.S. Nuclear Regulatory Commission Relating to the Certification of the AP1000 Standard Plant Design, Docket No. 52-006, SECY 05-0227* (accession number ML053630176). Washington D.C., January 24.

See Section 7.2

(Westinghouse 2005) Westinghouse Electric Company, LLC, *Design Control Document, Revision 15,* Appendix 1B, "Severe Accident Mitigation Design Alternatives," NRC Accession Number ML053460409, U.S. Nuclear Regulatory Commission, Washington, D.C., November 11, 2005.

See Section 7.2

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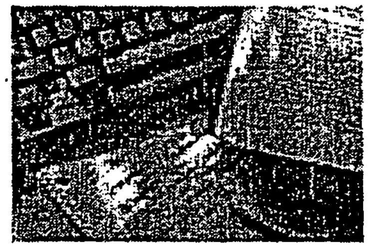

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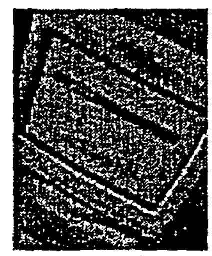

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ENVIRONMENTAL SURVEY OF TRANSPORTATION OF RADIOACTIVE MATERIALS TO AND FROM NUCLEAR POWER PLANTS

Atomic Energy Commission
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ENVIRONMENTAL SURVEY OF TRANSPORTATION OF RADIOACTIVE MATERIALS TO AND FROM NUCLEAR POWER PLANTS

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Directorate of Regulatory Standards

WASH-1238
UC-41

ENVIRONMENTAL SURVEY
OF TRANSPORTATION OF RADIOACTIVE MATERIALS
TO AND FROM NUCLEAR POWER PLANTS

(A general analysis of the impact on the environment of transporting radioactive materials to and from a light-water nuclear reactor in accordance with the regulatory standards and requirements of the Atomic Energy Commission and the Department of Transportation.)

Prepared by the
Directorate of Regulatory Standards
U. S. Atomic Energy Commission

December 1972

Foreword

This is a reissuance of an AEC Staff analysis of the environmental impact of the transportation of radioactive materials to and from nuclear power plants. It contains estimates of the potential exposures to transport workers and the general public under normal conditions of transport and the probabilities of occurrence and the potential consequences of accidents in transportation. This report, dated December 1972, was first issued in connection with the notice of a proposed amendment to 10 CFR Part 50, Appendix D, published in the Federal Register on February 5, 1973 to deal with environmental effects of the transportation of fuel and waste from nuclear power reactors, Docket No. RM 50-4. An informal public rule making hearing on the proposed amendment was held in Washington, D. C., on April 2, 1973. Minor corrections submitted during the hearing have been incorporated into this printing of the report.

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ENVIRONMENTAL SURVEY
OF TRANSPORTATION OF RADIOACTIVE MATERIALS
TO AND FROM NUCLEAR POWER PLANTS

(A general analysis of the impact on the environment of transporting radioactive materials to and from a light-water nuclear reactor in accordance with the regulatory standards and requirements of the Atomic Energy Commission and the Department of Transportation.)

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December 1972

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SUMMARY AND CONCLUSIONS

This analysis was made to assess the potential impact on the environment of transporting fuel and solid radioactive wastes for nuclear power plants under existing regulations. Most plants do not ship gaseous or liquid wastes off-site.

The regulations are based on two main considerations:

a) to protect the employees, transport workers and the public from external radiation in the transport of radioactive material under normal conditions, and

b) to assure that the packaging for radioactive materials is designed and constructed so that, under both normal and accident conditions, the radioactive material is unlikely to be released from the packaging.

The objectives of the first consideration are met by limitations on the radiation levels on the outside of packages of radioactive material and stowage and segregation provisions. Based on the detailed analysis which follows, we have estimated that the radiation dose under normal conditions of transport to the individual receiving the highest exposure is unlikely to be more than 500 mrem/yr and the average radiation dose to those individuals in the highest exposed group is about 100 mrem/yr. The Federal Radiation Council has recommended that the radiation doses from all sources of radiation other than natural background and medical exposures should be limited to 5000 millirem/year for individuals as a result of occupational exposure and should be limited to 500 millirem/year for individuals in the general population. The cumulative radiation dose to all transport workers is about 3 man-rem* per reactor year. The cumulative radiation dose to persons other than transport workers is about 2 man-rem per reactor year distributed among approximately 600,000 people. For purposes of comparison, the dose due to the average normal background radiation, about 130 mrem/person/year, would be about 78,000 man-rem per year for this group of 600,000.

The heat and weight in any one shipment and the total number of shipments from a typical light water reactor are small so there will be no appreciable effect on the environment from the shipping of the fuel and solid radwaste due to heat, weight, or traffic density.

Safety in radioactive material transport is achieved through design standards on packaging and implementation of a quality assurance program, including prooftesting and independent reviews, to assure conformance, to correct problems, and to help assure continued satisfactory (design) performance over the lifetime of the package under normal and accident conditions.

Every package must be designed and its use monitored to prevent release of radioactive materials not only during normal conditions of transport.

*Man-rem is an expression for the summation of whole-body doses to individuals in a group.

but also under other postulated abnormal circumstances developed through analyses and defined in the regulations.

The industry bears the primary responsibility for assuring safety in the packaging and transport of radioactive materials. The industry's activities are regulated by the Atomic Energy Commission (AEC) and the Department of Transportation (DOT). The regulatory functions include review of designs, quality assurance programs, testing, and use of packaging for radioactive materials.

The probability of an accident occurring in transportation is small, about one accident per million vehicle miles, and decreases with increased severity of the accident to about one severe accident per 100 million vehicle miles and one extremely severe accident per 10 million-million vehicle miles. For a typical nuclear power reactor, an estimated 112 shipments of fuel and wastes involving a total shipping distance of about 90,000 vehicle miles will be made each year. Based on these data, a shipment of fuel or waste will be involved in a transportation accident once in about 10 years and one accident out of about 100 will be severe. Because of the package design and quality assurance, the probability of a breach in the containment of a package involved in an accident is small and related to the accident severity. Because of regulatory limits on contents of packages and the nature and form of the unirradiated and irradiated nuclear fuel and solid radioactive waste from a light-water nuclear power plant, the amount of radioactivity which would be released if a breach were to occur in a package is unlikely to be large and although the consequences could be serious, they would not be catastrophic.

When both probability of occurrence and extent of the consequences are taken into account, the risk to the environment due to the radiological effects from transportation accidents is small. Accidents to packages more severe than the design basis accident for type B packages can occur, but the probability is very low (see Appendix A), and, although the consequences could be severe (see Appendix B), the risk is small. Because the risk from such events is so low and has been discussed in this Environmental Survey, evaluation of the environmental impact of such accidents would not be required of applicants in future Environmental Reports.

Within the United States over the past 25 years, there have been only about 300 reportable accidents in transportation involving packages of radioactive material. Only about 30% involved any release of contents or increased radiation levels, and none resulted in perceptible injury or death attributable to the radiation aspects. Millions of packages of radioactive material, including more than 3600 packages of irradiated fuel, have been transported during that period by all modes of transport.

The risk of injury or property damage from accidents due to common (i.e., other than radiological) causes in the transportation of nuclear fuel and solid radioactive waste also is small.

SECTION I. INTRODUCTION

A. Scope

In implementation of the National Environmental Policy Act of 1969, the AEC requires applicants for a license to operate light-water nuclear power plants to evaluate the environmental impact of transportation of nuclear fuel and solid radioactive wastes to and from the plant.

This is a general analysis of the impact on the environment from the transportation of nuclear fuel and solid radioactive wastes to and from a light-water-cooled nuclear power reactor in accordance with the present regulatory standards and requirements. The analysis is based on shipments of fresh fuel to and irradiated fuel and solid radioactive waste from a boiling water or a pressurized water reactor with design ratings in the range of 3,000 to 5,000 megawatts thermal (MWt) or 1,000 to 1,500 megawatts electrical (MWe). The nuclear fuel for the reactors considered was in the form of sintered uranium dioxide pellets encapsulated in zircaloy rods with a U-235 enrichment ranging from 1% to 4% by weight of the uranium present. The analysis was made with the assumption that present methods of transportation and existing standards and criteria for transportation will be applied over the operating life of the reactor.

Estimates were made of the impact from radiological effects and from common causes under normal conditions of transport and accidents. Transportation by truck, rail, and barge was analyzed, and probabilities of accidents calculated.

B. Purpose

This Environmental Survey dealing with the transportation of radioactive materials for nuclear power reactors under the present regulatory standards is being circulated as a "generic" analysis. It appears likely that the environmental impact of transportation from most nuclear power stations would fall within the scope of the parameters specified in this general analysis. It is anticipated that this "generic" analysis will provide the basis for the applicant's and the Commission's analysis of the impact on the environment of the transportation of fuel and solid radioactive waste under normal conditions of transportation and the design basis accident, i.e., accident damage test conditions specified in the regulations.

C. Principles of Safety in Transport

Most shipments of radioactive material move in routine commerce and on conventional transportation equipment. Shipments are therefore subject to the same transportation environment, including accidents, as non-radioactive cargo. Although a shipper may impose some conditions on the carriage of his shipment, such as speed limitations, providing an escort, etc., most of the conditions to which his shipment is subjected and the probability of his shipment being involved in an accident are not subject to his control. Protection of the public and transport workers from radiation during the shipment of radioactive materials is achieved by a combination of limitations on the contents according to the quantities and types of radioactivity and standards and criteria for package design and control. Safety in transportation does not depend on special routing, although special routings are used at some bridges and tunnels to avoid possible interference with the flow of traffic should an accident occur.

Primary reliance for safety in transport of radioactive material is placed on the packaging. The packaging must meet regulatory standards established by the Department of Transportation, Atomic Energy Commission and the States (see Section III) according to the type and form of material for containment, shielding, nuclear criticality safety, and heat dissipation. The standards provide that the packaging shall prevent the loss or dispersal of the radioactive contents, retain shielding efficiency, assure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified accident damage test conditions, (i.e., the design basis accident). The contents of packages not designed to withstand accidents are limited, thereby limiting the risk from releases which could occur in an accident. The contents of the package also must be limited so that the standards for external radiation levels, temperature, pressure, and containment are met.

Protection from external radiation is provided by limitations on the radiation levels on the outside of packages of radioactive materials and stowage and segregation provisions. The number of packages in a single vehicle or area is limited to control the aggregate radiation level and to provide nuclear criticality safety. Minimum separation distances from people and undeveloped film are specified for loading and storing packages of radioactive material to keep the exposure of persons and film to a minimum.

SECTION II. SUMMARY OF RESULTS OF THE DETAILED ANALYSES

A summary of the results of the analysis of the impact on the environment from transportation of fuel and solid radwaste associated with a light water nuclear power plant is given below. Details on each type of shipment are given in the Sections which follow. An analysis of accidents and some methods of calculations of doses and risks are presented in the Appendices.

A. Bases for Analyses

The estimates of the environmental effects of transportation are based on average conditions for such parameters as shipping distance, weather, radiation levels, package contents, population density, and accident frequency. The numbers of shipments of fuel and radwaste were estimated on the basis of those anticipated from a typical 1100 MWe light-water-cooled nuclear reactor. The degree of package damage assigned to different accidents represents judgment based on the results of tests of packages and the small number of accidents to date involving packages of radioactive material. The basis used for estimating the environmental effects is considered appropriate because, in the Staff's view, the effects are so small that further refinement is not warranted. If adjustment is desired for a particular case, suitable factors will be found in the details of the technical assessment.

The total number of shipments estimated to be shipped for a typical reactor each year are shown in Table 1, together with estimated average number of miles each type of shipment would be carried.

B. Heat

The amount of heat released from a shipment of unirradiated nuclear fuel or of solid radioactive waste is negligible. A rail cask containing irradiated fuel may release as much as 70 kilowatts of heat or about 250,000 Btu/hr. This might be compared to about 50 kilowatts of waste heat released from a 100 horsepower truck engine during full power operation. Even in those cases where more than one cask is located in an area, such as two or more loaded casks on a barge or train, the amount of heat released during shipment is too small to have any appreciable effect on the environment along the shipping route.

TABLE 1. SUMMARY OF INFORMATION ON SHIPMENTS

Type of Shipment	Mode of Transport	Estimated Weight (metric tons)	Heat Generated by Shipment (kilowatts)	Number of Shipments per 1100 MWe Reactor Year	Estimated Average Shipping Distance (miles)	Total Shipping Distance per Reactor Year (miles)
unirradiated fuel	truck	24	neg.	6* (18 initial)*	1000	12,000**
irradiated fuel	truck	35	10	60*	1000	120,000**
	rail	100	70	10*	1000	20,000**
	barge	150	140	5*	1000	10,000**
solid radioactive waste	truck	16	<.001	46	500	23,000
	rail	80	<.005	11	500	5,500

* plus an equal number of shipments for return of empty packagings

**only half of this distance involves shipments of radioactive material, the other half involves return of empty packagings.

The temperature on the accessible surface of packages in transport is limited by DOT regulations to 122°F if the package is shipped other than under "full load" conditions. Under "full load" conditions, the shipper has exclusive use of the vehicle and the cargo is loaded by the consigner and unloaded by the consignee so that contact with the package is controlled. Under "full load" conditions, the temperature on the accessible surface of the package is limited to 180°F. Under normal conditions of transport, there is unlikely to be damage to property or injury of persons due to external temperature.

C. Weight and Traffic Density

Shipments by truck must meet State restrictions on gross weight of vehicle which ensures against damage to bridges or roadways. The total number of shipments per reactor year, about 200, is too small to have any measurable effect on the environment due to the resultant increase in traffic density.

The weights of rail and barge shipments must meet the regulatory limitations of the Federal Railroad Administration and the U. S. Coast Guard and are within the range of weights of other commodities routinely handled on those modes of transport. The weights and numbers of shipments are too small to result in any measurable effects on the environment.

D. Radiation Exposures Under Normal Conditions

A summary of the estimated radiation exposures under normal conditions is given in Table 2. These estimates were based on average, realistic conditions as to radiation levels outside of packages, shipping distances, exposure time, distances from shipments, and numbers of people exposed. The details are given in the individual Sections which follow. The method of calculating the exposure of persons along the route is given in Appendix D.

The total impact on the environment from radiation in the transportation of fuel and wastes from a power reactor under normal conditions, based on the present packaging standards, is estimated to be a population dose of 5 man-rem per reactor year. An individual transport worker is unlikely to receive more than 500 mrem/yr. The average radiation dose to the highest exposed group of transport workers (truck drivers) is estimated to be about 100 mrem/yr. The cumulative dose to all transport workers is estimated to be about 3 man-rem per reactor year. The cumulative radiation dose to persons other than transport workers would be about 2 man-rem per reactor year, distributed among approximately 600,000 people. This is about one-millionth of the applicable Federal radiation

TABLE 2

ESTIMATED RADIATION DOSES
UNDER NORMAL CONDITIONS
PER REACTOR YEAR

Unirradiated fuel (by truck only)	Man-rem	Number of People
Transport workers	0.01	40
General public - onlookers	0.0003	60
- people along the route	0.001	3×10^5

Irradiated Fuel	Truck Man-rem	Truck No. People	Rail Man-rem	Rail No. People	Barge Man-rem	Barge No. People
Transport workers	1.2	4	0.05 (2.6)*	100 (22)*	0.04	10
General public - onlookers	0.8	600	0.1	100	-	-
people along the route	1	3×10^5	0.2	3×10^5	0.03	1×10^5

Solid Waste	Truck Man-rem	Truck No. People	Rail Man-rem	Rail No. People
Transport workers	1	4	0.05	100
General public - onlookers	0.6	500	0.1	100
people along the route	0.4	1.5×10^5	0.1	1.5×10^5

*For shipments transported by truck from the reactor site to a nearby railroad, transferred from truck to railroad car, and shipped by railroad car to the fuel recovery plant.

protection guide for the average exposure to the general population from all sources of radiation other than natural background and excluding radiation exposure for medical purposes. The dose to those same persons due to the average normal background radiation, about 130 mrem/person/year, would be about 78,000 man-rem per year.

E. Radiation Risk from Accidents

The risk of radioactive contamination or radiation exposure from accidents in transportation is extremely small.

As shown in the analysis of accidents in Appendix A, the probability of a truck, rail, or barge accident occurring in transportation is very small, about 10^{-6} per vehicle mile. Based on those accident statistics, the average number of shipments per year and average shipping distances, a shipment of nuclear fuel, solid radwaste, or empty fuel shipping containers for a typical nuclear power reactor would be involved in a transportation accident offsite about once for each 5 years of reactor operation.

More than 70% of the accidents which occur are of a minor nature and would produce little or no damage to a shipment. Less than 1% of the accidents involve a severe impact or fire.

The probability of a release of radioactive material or an increase in external radiation levels in an accident are small. One-third of the shipments are empty containers. In a severe accident, the vehicle may absorb most of the impact and the fire may not involve the shipment of radioactive material. Packages containing radioactive materials which might present serious potential radiation hazards if released must be designed to withstand accident conditions. The regulations limit the contents of packages not designed to withstand accident conditions, so only a small amount of radiation exposure would result should the package be severely damaged.

The extent to which the material is dispersed and the amount of radiation exposure that results from the release are affected by the weather conditions and the number of people in the vicinity of the accident (see Appendix B). The probability is small of a severe accident occurring in a location where the population density is high.

F. Risk from Common Causes

The impact on the environment from accidents in transportation of unirradiated fuel, irradiated fuel, solid radwastes, and empty containers due to common (non-radiological) causes is estimated to be 1 fatal injury in 100 reactor years, 1 non-fatal injury in 10 reactor years, and property damages of about $4[illegible]5 per reactor year (see Appendix C).

G. Alternatives

The risk of radioactive contamination or radiation exposure to the environment from the transportation of fuel and radwaste from a power reactor in containers designed to meet the present packaging standards is small. Alternatives and additional measures such as tightening of the standards to require additional accident protection and special routing of shipments, providing escorts, and requiring additional shielding in containers, were examined for the general case. Although some of the alternatives offer apparent advantages in terms of reducing the radiological effects on the environment, the overall risk from radiological effects is small. Any reduction in those effects by additional measures would to some extent be outweighed by an increase in adverse effects of a non-radiological character and by a large commitment of additional effort and equipment. Adoption of one or more of the alternatives in specific cases might be justified. However, the advantages of the alternatives do not appear to be sufficient to warrant their adoption as general requirements.

SECTION III. REGULATORY STANDARDS AND REQUIREMENTS

Packaging and transport of radioactive materials are regulated at the Federal level by the Atomic Energy Commission,[1] the Department of Transportation (DOT),[2] and the U. S. Postal Service.[3] Certain aspects, such as limitations on gross weight of trucks and transportation not subject to DOT, AEC, or the Postal Service regulations, are regulated by the States. Most States have adopted regulations pertaining to intrastate transportation of radioactive materials which require the shipper to conform to the packaging, labeling, and marking requirements of the U. S. Department of Transportation to the same extent as if the transportation were subject to the rules and regulations of that agency.

A. Packaging Standards and Requirements

The packaging standards and criteria are found in the regulations of the AEC (10 CFR Part 71) and the regulations of the DOT (49 CFR Parts 170 through 179).

The present criteria provide assurance that packaging designed to meet such standards can be carried on all modes of transport and will withstand the conditions likely to be encountered in accidents. As developed, the criteria specify tests of packaging which can be carried out either in the laboratory or in the field with conventional and readily available equipment and facilities. The criteria, which were first published by the International Atomic Energy Agency in 1964, have been adopted in many international and national transportation regulations and served as the basis for the regulatory standards and criteria of the U. S. They were based on a detailed analysis of normal and accident conditions in transport and nearly 20 years of experience in shipping many types of radioactive materials.

To meet the regulatory standards, packaging must be designed and constructed to provide two and, in some cases, three levels of protection.

The packaging must function in the normal transportation environment with a high degree of reliability. Systems selected to achieve the basic design functions, i.e., containment, shielding, heat dissipation, and nuclear criticality safety, must provide a high degree of inherent safety under normal conditions and have a high tolerance for malfunctions, off-normal conditions, and accidents should they occur. Each shipping container is checked routinely to assure that the "as built" high quality is maintained throughout its lifetime.

Despite the best possible design practices and the highly assured capability for reliable and practicable operation, allowance is made for malfunctions, off-normal conditions, and accidents by providing an additional level of protection to resist or accommodate such occurrences. As with the primary level of protection, conservative design practices, adequate safety margins, and inspectability are incorporated into these secondary protection systems to assure both the effectiveness and reliability of the second level of defense. In addition, these systems are designed to be routinely examined and tested so that there is full assurance that they will operate reliably if required.

As an added measure of safety, where the design includes mechanical systems essential to safety, the design is evaluated under normal conditions and against a series of severe hypothetical accident conditions, assuming certain of these protective systems fail. If such failure could produce serious consequences, additional protective measures, or redundancy of the safety system must be provided.

TABLE 3. QUANTITY LIMITS AS RELATED TO PACKAGE REQUIREMENTS

Transport Group	Examples	Exempt Quantity (curies)	Type A Package (curies)	Type B* Package (curies)
I	^{239}Pu, ^{242}Cm, ^{252}Cf	10^{-5}	10^{-3}	20
II	^{210}Bi, ^{90}Sr, ^{210}Po	10^{-4}	5×10^{-2}	20
III	^{137}Cs, ^{192}Ir, ^{131}Ir	10^{-3}	3	200
IV	^{76}As, ^{14}C, ^{45}Ca	10^{-3}	20	200
V	Noble gasses, ^{85}Kr	10^{-3}	20	5,000
VI	^{37}Ar, ^{133}Xe, ^{85}Kr uncompressed	10^{-3}	1,000	50,000
VII	Tritium - as a gas or in luminous paint	25	1,000	50,000
Special Form	Co^{60} radiography source, Pu-Be neutron source	10^{-3}	20	5,000

*A Large Quantity is defined as any quantity in excess of a Type B quantity.

The type of packaging is specified in DOT regulations, 49 CFR 173, according to the type and quantity of radioactive material (see Table 3).

Radioactive materials are divided into two broad classes. (1) "special form" which is a massive, non-friable, solid material or material confined in a high integrity capsule of inert material, and (2) "normal form" which applies to all radioactive materials which are not "special form." Normal form radioactive materials are classified into seven groups of radionuclides based primarily on radiotoxicity of the radionuclides. Package limits for the seven transport groups and "special form" are shown in Table 3.

Small quantities of radioactive materials, certain concentrations, small quantities of radioactive materials in manufactured goods, and low specific activity materials may be shipped in strong industrial packages and are exempt from specification packaging, marking and labeling with the radioactive material label. The Postal Service regulations generally allow the exempt quantities to be shipped by mail in leakproof containers.

Type A quantities of radioactive materials must be shipped in packaging, identified as Type A packaging, which will prevent loss or dispersal of the radioactive contents and retain shielding efficiency and effectiveness of other safety features under normal conditions of transport. Standards for evaluation and testing of adequacy with respect to normal conditions specified in AEC and DOT regulations include temperatures ranging from -40°F to 130°F, all surfaces except the bottom wet for 30 minutes, being subjected while wet to a 4 foot free fall, vibration normally encountered in transport and external pressure reduced to 0.5 atmosphere.

Quantities exceeding Type A quantities must be shipped in Type B packaging. Type B packaging must be designed to withstand normal transport conditions without loss of contents or shielding efficiency and to suffer no more than a specified loss of contents or shielding efficiency if subjected to a specified sequence of accident damage test conditions. That damage test sequence includes: (1) a free fall from a height of 30 feet onto an unyielding surface with the package landing in the orientation which does the most damage, (2) a free fall from a height of 4 feet onto a 6-inch-diameter steel plunger long enough, and with the package in the orientation, to do maximum damage, (3) heat input from exposure for 30 minutes to a fire or other radiant environment having a temperature of 1475°F and an emissivity of 0.9, and (4) for fissile material, immersion in water to a depth of 3 feet for 24 hours. Those test conditions make up the design basis accident for type B packages; i.e., package designs which meet the criteria under these test conditions are considered to provide adequate protection to the public and operating personnel in transportation accidents.

Large quantities must be shipped in Type B packaging which provides for adequate dissipation of heat. In addition, there must be no loss of contents at an external pressure of 25 psig, which is approximately equivalent to immersion in water to a depth of 50 feet.

With respect to heat dissipation, the regulations require the package to be designed so that the temperature rise due to decay heat will not adversely affect the package or the contents and will not cause excessive pressure. The accessible surface of the package must not exceed a temperature of 180°F.

B. Nuclear Criticality Safety

Fissile material (i.e., uranium-233, uranium-235 and plutonium) in quantities exceeding 15 grams per package or, in homogeneous, hydrogenous solutions and mixtures, quantities exceeding 500 grams of U-233 or Pu or 800 grams of U-235 per package, require some control in transport to assure safety from accidental criticality. Nuclear criticality safety in transport is provided by assuring that the contents of each package of fissile material is subcritical when delivered to a carrier for transport and that the package is so designed that it will remain subcritical under all conditions likely to be encountered in transport, including accidents. In addition, the contents must be limited or the package must be designed so that the number of packages which are likely to be accumulated in one vehicle or area will be subcritical under all conditions likely to be encountered in transport, including accidents and handling errors.

The AEC regulations specify the conditions for evaluating the adequacy of design of a package for fissile material including form and geometry of the contents and moderation and reflection.

The package design must be evaluated against the accident damage test conditions discussed earlier for Type B packages.

A package for fissile material must be so designed and constructed and its contents so limited that the following numbers of such packages can be shown to be subcritical in a moderated and reflected array according to the Fissile Class (I, II, or III) to which the package is assigned.

	Normal Conditions	All Packages Damaged as in Accident Conditions
Fissile Class I	any number	250 packages
Fissile Class II	5 times the allowable number*	2 times the allowable number*
Fissile Class III	beside an identical shipment	the allowable number*

* The allowable number is the number of the same type of packages to be allowed in one shipment.

The conditions for transport vary according to the Fissile Class. Fissile Class II packages are controlled by the carrier as to an allowable number on a vehicle or in one handling or storage area. This is done by the simple system of assigning a number to each package, called a transport index, and requiring the carrier not to allow more than an accumulation of 50 transport indexes on a vehicle or area. This system has been applied to limiting the accumulated radiation level since 1948.

For Fissile Class III, the shipment must be made exclusive use (i.e., the consignor loads the shipment and the consignee unloads the shipment and nothing is allowed on the vehicle other than the consignor's material) or by an escort provided by the shipper who assures the shipment is kept separated from other fissile material, or some other procedure specifically approved by DOT.

Fissile Class I packages do not require limitations on the number of packages in an area or vehicle for nuclear criticality safety.

In some cases physical properties limit the number of packages in a shipment. For example, in most cases one irradiated fuel cask is shipped on a truck or rail car and the cask is shipped exclusive use because of weight limitations on the vehicle even though some designs might meet the Fissile Class I requirements. For unirradiated nuclear fuel, the allowable number of packages for Fissile Class II in the case of one design of PWR package is 20. However, because of the size and weight of each package, only 6 or 7 can be loaded on one truck.

C. Packaging Design Review

At the present time, the AEC reviews and issues approvals for designs of packages for shipping large quantities and fissile materials. DOT reviews and issues approvals for Type B package designs and, based on AEC evaluations, issues approvals for large quantity and fissile material package designs.

Applicants for approval of a packaging design must provide a detailed analysis of that design to demonstrate that the design meets the packaging standards and criteria. The demonstration that the packaging design is adequate may be made by quantitative assessment, tests of models or packaging details or mock-ups representing the methods of construction used, extrapolation from test results for similar designs or designs employing similar construction features, actual tests of samples of packaging made to the design, or other evidence.

The DOT plans to discontinue issuing specific approvals for radioactive material packages which meet all of the packaging standards. In December 1971, the AEC and DOT published[4] proposed regulatory changes under which DOT would transfer to the AEC all of the radioactive material packaging approval functions. The final regulatory changes are expected to be published within the next few months.

D. Quality Assurance and Control

It is possible that a package will be constructed or used in a manner not in accordance with the design; however, the likelihood of such errors is considered small in view of the regulatory requirements for quality assurance and for various observations and tests before each shipment.[5]

Under the Department of Transportation regulations, each fabricator of specification containers must register with, and is subject to inspection by, DOT.

The regulations specify certain tests that must be carried out on such containers. Under AEC regulations, licensees who wish to fabricate casks are asked to describe their quality assurance program when they apply for approval of the design. In addition, the regulations require that packages for fissile material and large quantities be tested prior to first use with respect to shielding and heat dissipation and prior to each use as to proper assembly, proper closing, temperature, pressure, and presence of neutron absorbers.

E. Radiation Level Limitations

External radiation exposure of transport workers and the general public in the transportation of packages of radioactive material is controlled during transport by several different methods.

The radiation emitted from individual packages of radioactive material is limited by the DOT regulations[6] to no more than 200 mrem/hr on the surface to limit the direct exposure to the person handling the package, and no more than 10 mrem/hr at 3 feet from the surface of the package to limit the radiation level to which persons and property in the vicinity of the package would be exposed.

If a package is shipped in a closed truck or rail car under the "exclusive use" conditions (which means it is loaded by the consignor and unloaded by the consignee), the radiation level at 3 feet from the surface of the package is limited to 1000 mrem/hr provided the radiation level does not exceed 200 mrem/hr at the surface of the vehicle, 10 mrem/hr at 6 feet from the outside surfaces of the vehicle, and 2 mrem/hr in either the driver's compartment or other normally occupied positions in the truck or rail car.

As a simple indicator of the radiation dose rate from an individual package, the regulations define one "transport index" (TI) as being equal to 1 mrem/hr at 3 feet from the surface of the package. The regulations specify limits for aggregations of packages in terms of the sum of the transport indexes. The number of packages stored or handled in one area or loaded on one car or vehicle must be so limited that the sum of their transport indexes does not exceed 50. This prevents a large aggregation of packages, each with a significant radiation level, from producing a much higher radiation level than desirable because of the additive effect of the radiation levels from all of the packages.

Simple tables of minimum separation distances[7] from people and unexposed film are specified for packages of radioactive materials in storage and on vehicles in terms of the sum of the transport indexes in each group of packages.

Whether there is one package or a large number of packages in a vehicle or a location, the transport worker or carrier is required to read each TI, add the total number of TI's present, determine from the tables in the regulations the distance those packages must be kept from film and continuously occupied areas, and assure that those separation distances are provided.

The transport index system has also been adapted for limiting aggregations of packages containing fissile radioactive materials to assure nuclear criticality safety. The shipper determines in accordance with specific criteria laid down in the AEC regulations of the AEC a transport index figure which is to be assigned to the fissile material package. For shipping, the shipper assigns to each package of fissile material the nuclear safety TI as calculated or the radiation level TI (as described earlier), whichever is the higher. The transport worker, as is the case for radiation levels, adds the TI's and by complying with the limitation on the number of TI's in any one vehicle or location limits the amount of fissile material in all types of packages to safe limits. The TI assigned to individual packages of fissile material for nuclear safety reasons takes into account that, in cases other than exclusive-use shipments, 2 times, or as many as 5 times the permitted total number of TI's in a collection of packages may be inadvertently placed together.

It will be recognized that mixing nuclear safety TI's with radiation level TI's in the course of transport increases the margin of safety for both since they are not synergistic.

F. Surface Contamination Levels

DOT regulations[8] also require that there be no significant removable surface contamination on the external accessible surfaces of packages when they are shipped. Levels of removable contamination on the surfaces are determined by a wipe test. The regulations consider the level is "not significant" if the activity on the wipe does not exceed 10^{-11} Ci/cm^2 for beta-gamma emitters and 10^{-12} Ci/cm^2 for alpha emitters. Any fixed contamination of the surface is limited by the external radiation level limitations discussed in the previous paragraphs.

G. External Temperature

The DOT regulations[9] limit the temperature at any accessible surface of the cask to not more than 122°F at any time during transport, except that for full load or exclusive use shipments, the temperature may be 180°F.

H. Warning Labels

Each package of radioactive material is required by DOT regulations[10] to be labeled on two opposite sides with a distinctive warning label. Each of three label formats bears the unique trefoil radiation symbol. The label alerts persons handling packages that the package may require special handling. If the background color of the label is all white, the radiation is minimal and nothing special is required for that package. If, however, the background of the upper half of the label is yellow, a radiation level requiring consideration may exist at the outside of the package, and an indication of what controls must be exercised for that package is related to the transport index concept discussed above. If the package bears a yellow label with three stripes, the rail or highway vehicle in which it is carried must be placarded.

I. Placards

A truck or rail car carrying any package labeled with a Radioactive Yellow-III label must be placarded on the outside.[11] The placard for rail cars bears the distinctive trefoil symbol and, for trucks, the word RADIOACTIVE in letters large enough to catch the eye. The principal purposes of placards are to advise freight handlers of the presence of radioactive material with TI's inside the vehicle, or to indicate the presence of special types of shipments (e.g., a Fissile Class III package, a special permit package, or a large source package); and to warn passers-by and emergency crews that radioactive material shipments are in the vehicle. This marking or placarding is intended to encourage persons not to remain in the vicinity of the vehicle unnecessarily so as to reduce exposures which would otherwise result from loitering in the vicinity. Also, the placard will alert emergency crews to the need for taking appropriate precautions in case such vehicles are involved in accidents. Cars and trucks carrying carload or truckload lots of radioactive materials, packages with significant external radiation levels or containing large quantities of radioactive material, or Fissile Class III shipments are required to be placarded with a "Radioactive" placard.

J. Capacity for Coping with Accidental Releases

The consequences of an accident involving radioactive material are mitigated by the procedures which carriers are required to follow.[12] These procedures include: segregation of packages and materials from persons; immediate notification of the shipper and DOT in case of an accident, fire, or leaking package; and a requirement that vehicles, cars, building areas, and equipment not be placed in service again until surveyed and, where necessary, decontaminated.

Trained personnel equipped to monitor the area and competent to act as advisers are available through an inter-Governmental radiological assistance program. The radiological assistance teams are dispatched in response to calls for emergency assistance. This assistance has been made available in the few transportation accidents involving radioactive materials shipments which have occurred in recent years. Should a major release occur, this type of assistance might help reduce the impact of the release.

K. Shipper's Certification

Before delivering a package to a carrier for transport, the shipper must determine that there is no "significant" loose radioactive contamination on the outside of the package, that the radiation levels on the surface of the package and at 3 feet from the package meet the specified regulatory levels, and that the marking and labeling are in compliance with the requirements. The shipper also must certify[13] in writing on the shipping papers that the radioactive materials are properly classified, described, packaged, marked, and labeled and are in proper conditions for transport according to the applicable regulations of the Department of Transportation.

L. Weight and Traffic Density

State highway weight restrictions limit the gross weight of trucks for routine shipments so that the gross weight of casks are limited to about 25 tons. Shipments of casks weighing up to about 35 tons may be allowed in most States under a special overweight permit. The States often prescribe special routing for overweight shipments and in some cases restrict the period during which the truck can travel.

Repetitive shipments of overweight loads may cause breakup of the roadway. Some irradiated fuel shipping casks may require overweight permits. The number of such shipments is limited to about 60 round trips per year per reactor. That number of overweight shipments would not be expected to have any adverse effect on the roadways. Rail shipments of 50 to 100 tons of other commodities, such as coal, are routinely handled, so rail shipments of casks of comparable weights would offer no unusual loading for rail facilities. Barges also routinely transport cargoes weighing more than 100 tons.

With respect to traffic density, the average number of truck shipments of nuclear fuel, solid rad-waste, and empty packagings is estimated to be about 200 per year for a typical reactor and involves a total of about 155,000 truck miles. The number of shipments and miles travelled are small compared to the present traffic densities and miles travelled by trucks for all purposes.

As an indication of the traffic flow, an average of 43,500 motor vehicles per day traveled over one section of I-5 between San Diego and Los Angeles in 1971. According to the Federal Highway Administration, the average number of trucks per day on any given section of U. S. highway generally varies from about 100 to 10,000. The total number of truck miles traveled in 1971 is estimated to be over 12 billion.

M. Changing the Standards and Requirements

The safety of radioactive material transport is assured not only through the design standards for packaging but also by quality assurance programs to assure conformance with approved designs, to correct problems and to help assure continuing satisfactory performance over the lifetime of the package. Despite use of the best possible design practices, assurance of the capability for reliable and predictable operations of the packaging and the transportation equipment, employing measures to reduce the already low probability of accidents, and provisions to mitigate the consequences of accidents which may occur, errors, malfunctions, off-normal conditions, and accidents will occur. Such accidents are required to be reported and will be investigated. If as a result of such events, data and experience associated with the changing characteristics and increased numbers of shipments of radioactive material, or changes in the useful life of the equipment or in the transportation methods, evidence becomes available that accepted guidelines are being exceeded or the public is being unduly exposed or their health and safety impaired, action can and will be taken to correct the causes in a timely manner. The regulatory requirements, codes, standards, specifications and criteria applicable to the designs of packages, loading patterns, protective measures, and quality assurance practices for the transportation of radioactive material can be modified should the need for changes become evident.

The probability of leakage due to human error can be reduced by increased control over the preparation of packages for shipment. Two actions already are underway which are intended to increase that control. DOT recently amended its regulations[14] to require that shippers carry out certain examinations and test procedures on packages prior to shipment. The Atomic Energy Commission is considering expanding its quality assurance requirements applicable to packages used by its licensee-shippers.

IV. DETAILED ANALYSIS OF THE ENVIRONMENTAL IMPACT OF TRANSPORTING UNIRRADIATED FUEL TO A TYPICAL LIGHT-WATER NUCLEAR REACTOR IN ACCORDANCE WITH PRESENT REGULATORY STANDARDS AND REQUIREMENTS

A. Characteristics

The nuclear fuel for an 1,100 MWe reactor typically consists of 100 metric tons (MT) of uranium for a pressurized water reactor or 150 MT for a boiling water reactor. The uranium enrichment varies from about 1% to 4% U-235 by weight depending on the reactor design. The fuel is in the form of uranium dioxide which has been sintered and compacted to form very dense, high-strength, high-melting-point pellets approximately 1-1/4 centimeters (cm) in diameter and 2 cm in length. The pellets are stacked in zircaloy tubing which is welded shut at both ends to form a fuel rod. The fuel rods are subjected to rigorous quality control to ensure their integrity prior to use in the reactor. A fuel element is made up of 50 to 200 fuel rods about 4 meters (m) long, weighs from 250 to 700 kilograms (kg) and contains approximately 200 kg of uranium for a BWR or 500 kg of uranium for a PWR.

About one-third of the fuel in a PWR or about 1/5 of the fuel in a BWR, i.e., about 30 MT of fuel, is replaced each year. Unirradiated fuel (also referred to as cold or fresh fuel) is shipped by truck, usually two fuel elements per package, in long packages, 16 packages of BWR elements or six packages of PWR elements constituting a truckload. About six truckloads of fuel elements are shipped to a reactor each year.

B. Packaging

As indicated in the introduction, the packaging provides much of the assurance of safety in transport of radioactive materials. The design of the packaging for shipment of unirradiated fuel, the contents, the transport index to be assigned each package (if Fissile Class II), and any special procedures to be followed in loading the fuel into the package and closing the package must meet standards set forth in AEC regulations.[1] Each package design must be reviewed and approved by the AEC prior to first use. Labeling of the package and other transport conditions are specified in DOT regulations.[2]

The packaging must ensure against nuclear criticality under both the normal conditions of transport and accident damage test conditions and prevent loss of contents under normal conditions of transport.

The fuel elements are usually enclosed in a plastic bag and placed in a metal container which supports the fuel element along its entire length during the course of transportation. A typical shipping container for PWR fuel elements is a cradle assembly consisting of a rigid beam or "strongback" and a clamping assembly which holds the fuel elements firmly to the "strongback." The "strongback" is shock-mounted to a steel outer shell by shear mounts. BWR fuel elements are shipped in steel boxes which are positioned in an outer wooden box by cushioning material. Packaging for PWR fuel elements is cylindrical in shape, approximately 1.2 m in diameter and 4.9 m long and ranges in weight when loaded from 2800 to 4000 kg. Packaging for BWR fuel elements is rectangular in shape, approximately 1 m high, 1 m wide, and 5.2 m long. When loaded, the package weighs up to 1300 kg. Examples of types of shipping containers are shown in Figures 1, 2, and 3.

C. Transport Conditions

Almost all shipments of unirradiated fuel are now, and will continue to be, made by truck. Rail shipments take too long, and many nuclear power plants do not have rail facilities. Water shipments take even longer, and there are very few convenient barge routes between the fuel fabricators and the nuclear power plants. Shipments by air are also unlikely, in spite of the short transit time. The packages are long (about 5 m), freight rates are high, and most reactors are some distance from the major airport facilities having cargo aircraft.

It will require about 18 truckloads of fuel to load the reactor initially; thereafter, about six truckload shipments of fuel will be required annually for refueling. Each shipment will travel a distance of about 1000 miles on the average, (a minimum distance of 25 miles to a maximum of 3,000 miles).

In most cases, a shipment of unirradiated fuel will be transported by exclusive use. i.e., as a "full load." The packages would be loaded on the truck at the fuel fabrication plant by the shipper, transported by the carrier directly to the nuclear power plant and unloaded by the power plant personnel, with no intermediate off-loading, storage, or intervehicular transfers enroute. No other shipments would be loaded on the vehicle except by the shipper himself. Average transit time will be about 3 days, based on present experience.

FIGURE 1

BWR FUEL ELEMENT SHIPPING CONTAINER

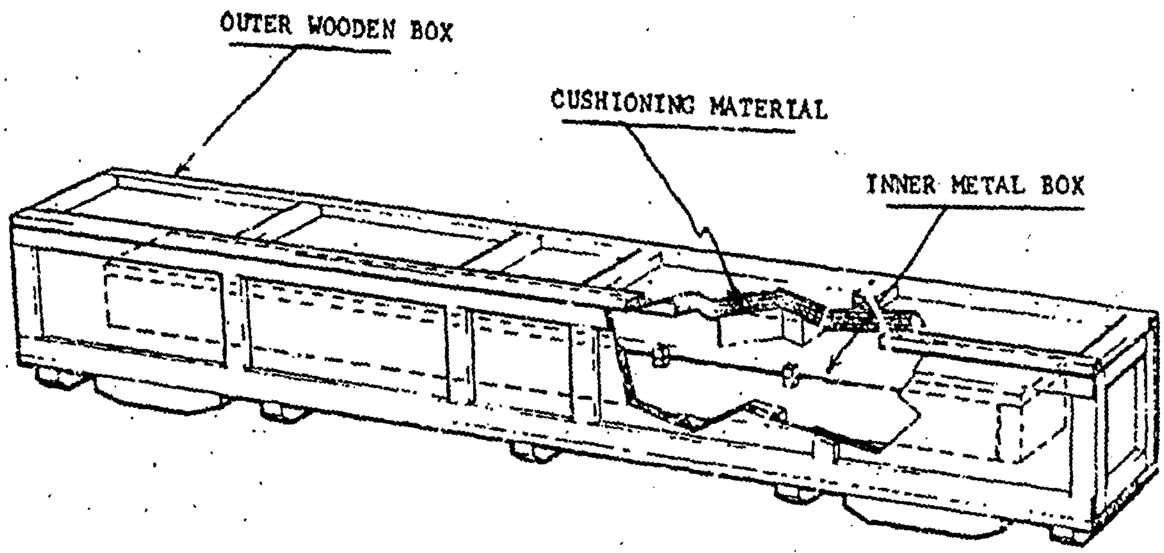
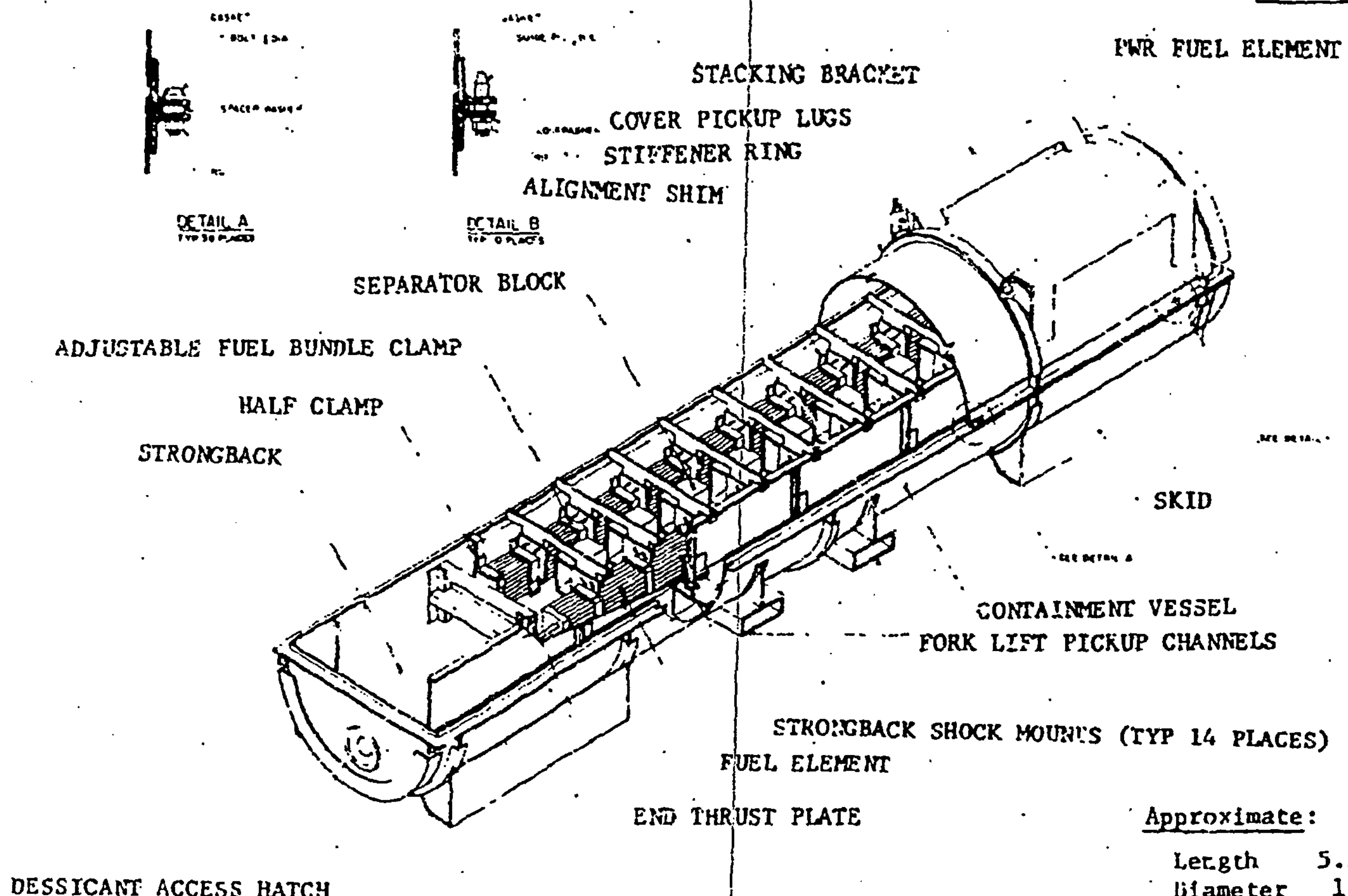


FIGURE 2

PWR FUEL ELEMENT SHIPPING CONTAINER



Approximate:

Length	5.5 meters
Diameter	1 meter
Weight	
empty	2 tons
loaded	3.7 tons

Figure 3 TRUCKLOAD OF COLD FUEL

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Some shipments will be made "less-than-truckload" (LTL) i.e., one or two packages shipped via general freight or express and transported in a vehicle along with other freight, in accordance with the DOT regulations. The packages would be moved from truck to truck, through terminals and "in transit" storage. The average transit time will be about 5 days, based on present experience.

D. Effects on the Environment

Normal Conditions

1. Heat

In the case of unirradiated fuel, there will be no readily detectable heat output.

2. Weight and Traffic Density

The number of shipments of unirradiated fuel will average about 6 truckloads per year. The total number of such shipments is too small to have a measurable effect on the environment due to the resultant increase in traffic density.

The number of packages per vehicle can be adjusted so that the transporting vehicle can stay within the cargo gross weight limitations of the State (usually about 25 tons); hence, there would be no excessive load on the roadbeds or bridges for major routes.

3. Radiation

a. External radiation exposure levels

The radioactivity in a package of unirradiated fuel will be about 0.5 to 2.0 curies. Based on data obtained from AEC licensees and contractors, the radiation level at the surface of the unirradiated fuel containers is likely to average about 1 millirem per hour (mrem/hr). For an individual package the radiation level at 3 feet from the surface of the package would be about 0.4 mrem/hr, and at 15 feet about 0.05 mrem/hr. For a cluster of six to 16 packages, the radiation levels would be about 1.5 mrem/hr at the edge of the cluster, 0.7 mrem/hr at 3 feet, and 0.06 mrem/hr at 15 feet. The radiation level at the outside surface of a truck containing a load of unirradiated fuel would be about 1 mrem/hr, and at 3 feet from the surface of the truck about 0.1 mrem/hr. From actual experience, the level in the cab of a truck would be about 0.01 mrem/hr above the natural background.

(1) Truck Drivers. Two truck drivers during a 1000 mile trip would probably spend no more than 20 hours in the cab and about one hour outside the truck at an average distance of 3 feet from the cargo compartment. Under those conditions, each driver could receive about 0.3 mrem/shipment or about 1.8 mrem/yr for six shipments. The cumulative annual dose to all drivers would be about 0.004 man-rem.*

(2) Freight Handlers. For shipments which are transported as "full-loads," exposure to the carriers' freight handlers would be zero, since the packages are loaded by fuel fabrication personnel and unloaded by personnel at the reactor and are not handled enroute.

For less-than-full-load shipments, the packages may spend an average of about 12 hours on loading docks and 24 hours in storage. While in storage, the exposure to handlers is essentially zero.

Handling on the docks requires mechanical equipment because of the weight of the packages. Based on actual handling experience, for about 11 of the 12 hours, the packages would be in relatively isolated "route grouping" areas, with an average of from one to three handlers being exposed a total of from 10 minutes to 1 hour each, from an average distance of 3 to 15 feet. The exposures would then range from 0.01 mrem to 3 mrem per handler. The average exposure would probably be in the range of 0.2 mrem per handler for three handlers, or about 0.004 man-rem per year for six shipments.

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*Man-rem is an expression for the summation of whole-body doses to individuals in a group. In some cases, the dose may be fairly uniform and received by only a few persons (e.g., drivers and brakemen); in other cases, the dose may vary and be received by a large number of people (e.g., persons along the shipping route).

During actual transfer of a shipment, three handlers (not necessarily the same three handlers previously mentioned) would probably not be exposed for more than 30 minutes each, at an average distance of 3 feet, each receiving about 0.25 mrem. That would be a total annual exposure of about 0.005 man-rem.

(3) General Public--Onlookers. Members of the general public are normally excluded from loading and unloading operations, but exposures might occur at enroute truck stops for fuel and eating. Trucks are placarded on both sides and the front and rear as "Radioactive." Members of the general public are unlikely to remain near a truck more than a few minutes. If a person spends 3 minutes at an average distance of 3 feet from the truck, the dose would be about 0.005 mrem. If 10 persons, on the average, were so exposed, the total annual dose to such onlookers would be about 0.0003 man-rem.

(4) General Public--Along the Shipping Route. The radiation level at 6 feet from a vehicle loaded with packages of unirradiated fuel will likely be no more than 0.1 mrem/hr. Consider the vehicle travels 200 miles per day, and the mean population density along the route is 330 persons per square mile. For a trip of 1000 miles one way and 6 trips per year, the cumulative annual dose to approximately 300,000 persons in an area along that route between 100 feet and 1/2 mile on either side of the vehicle would be about 0.001 man-rem. See Appendix D for detailed calculation.

(5) Animals. The exposure of domestic animals or pets during transit might occur during terminal transfers of unirradiated fuel packages. If such exposures did occur, the average would probably be about the same as for freight handlers, i.e., about 0.3 mrem each.

(6) Film. Unexposed photographic film can be affected by radiation and is the most radiation-sensitive material likely to be transported together with radioactive materials.

Under full-load conditions, film would not be shipped with unirradiated fuel. Under LTL conditions, DOT regulations require film to be separated by at least 15 feet from shipments of radioactive material. A shipment of film within 15 feet of a shipment of cold fuel for 12 hours would receive an exposure of about 0.6 mrem. This would not produce any measurable effect on the film.

Accident Conditions

1. In-Plant Accidents

The "in-plant" radiological aspects of transportation of radioactive material are evaluated separately as part of the licensing procedures or contractual requirements and are not evaluated against the packaging standards and criteria for transportation. For that reason, the "in-plant" aspects have not been included in this analysis.

2. Offsite Accidents

A truckload of unirradiated fuel from a typical reactor may be involved in an accident about once in 110 reactor years (see Appendix A). The packages are so designed that in the unlikely event a shipment of unirradiated fuel is involved in an accident, it is unlikely the fuel will be released.

The fuel rod is constructed to withstand internal and external pressures, from 1000 to 2000 pounds per square inch gauge, anticipated in operation of the reactor. Its construction is such that release of the pellets of uranium oxide or the oxide itself is unlikely. Fuel rods of this type have been tested by being dropped 30 feet onto concrete on end, on the side, and at an angle of 45°, without rupture of the cladding or loss of contents.[15]

The pelletized form of the uranium and its encapsulation make releases of radioactivity in an accident extremely unlikely. Because of the low specific activity of the fuel, the radiation level associated with the fuel itself is quite low. Therefore, except for an accident resulting in nuclear criticality, the radiological impact on the environment from accidents involving unirradiated fuel is negligible.

The packaging is designed to prevent criticality under normal and severe accident conditions. An accident which could lead to accidental criticality would require release of several fuel elements as a result of severe damage or destruction of more than one package, which is unlikely to happen other than in an extremely severe accident. After release from the packages some of the fuel elements must be assembled in a close array and moderated, for example, by being submerged in water; accidental criticality in air is not possible. Considering the requirements for package design and controls exercised over packages during transport, the probability of such an accident is so small that, in practice, it is considered to be incredible.

Based on the above, the impact on the environment from radiation in transportation accidents involving unirradiated fuel is considered to be negligible.

V. DETAILED ANALYSIS OF THE ENVIRONMENTAL IMPACT OF TRANSPORTING IRRADIATED FUEL FROM A TYPICAL LIGHT-WATER NUCLEAR REACTOR IN ACCORDANCE WITH PRESENT REGULATORY STANDARDS AND REQUIREMENTS

A. Characteristics

Each year, on the average, from one-fifth to one-third of the fuel in a reactor is replaced with fresh fuel. A fuel element removed from the reactor will be essentially unchanged in appearance and will contain some of the original useful uranium-235, which is recoverable. On the average, the fuel will have been irradiated to 33,000 megawatt-days per metric ton (MWD/MT). As a result of the irradiation and fissioning of the uranium, the fuel element will contain some plutonium and large amounts of fission products. As the radioactive atoms decay, they produce radiation and decay heat. The amount of radioactivity remaining in the fuel varies according to the length of time after discharge from the reactor. After discharge from the reactor, the fuel elements are placed under water in a storage pool for radioactive decay and cooling prior to being loaded into a cask for transport.

The amount of radioactivity in the spent fuel decreases quite rapidly during the first few days after discharge. After 150 days cooling, however, each irradiated PWR fuel element still contains approximately 2,000,000 curies of radioactivity, of which 5,000 curies is in gaseous form (see Tables 4 and 5). The radioactivity in a BWR element is about half those values.

B. Packaging

Packaging for the shipment of irradiated fuel, called casks, must meet the DOT and AEC regulatory requirements for fissile material packages and for large quantity packages; that is, casks must ensure against nuclear criticality and loss of contents under normal conditions of transport and under accident damage test conditions, provide shielding to reduce the radiation emitted from the cask to specified levels, and dissipate the heat generated in the fuel and cask by radioactive decay. At present, there is only one approved design for a cask which has sufficient length, cavity diameter, shielding, and heat dissipating capacity to be used for transporting the forthcoming generation of irradiated fuel assemblies from nuclear power reactors. Other proposed designs of casks for such fuels which the applicants consider meet the regulations are currently being reviewed by the AEC.

TABLE 4

Radioactivity of Irradiated Fuel[16]
(curies per metric ton of uranium)

	Cooling Period (in days)			
	90	150	365	3650
Fission Products	6.19×10^6	4.39×10^6	2.22×10^6	3.17×10^5
Actinides (Pu, Cm, Am, etc.)	1.42×10^5	1.36×10^5	1.24×10^5	
Total	6.33×10^6	4.53×10^6	2.34×10^6	

TABLE 5

Predominant Fission Products in Gaseous Form[16]
Included in Radioactivity of Irradiated Fuel
(curies per metric ton of Uranium)

	Cooling Period (in days)			
	90	150	365	3650
Krypton-85	1.13×10^4	1.12×10^4	1.08×10^4	6.05×10^3
Xenon-131m	1.06×10^2	3.27	1.08×10^{-5}	
Iodine-131	3.81×10^2	2.17	1.98×10^{-8}	

TABLE 6

Thermal Energy in Irradiated Fuel[16]
(watts per metric ton of uranium)

	Cooling Period (in days)			
	90	150	365	3650
Thermal Energy	2.71×10^4	2.01×10^4	1.04×10^4	1.06×10^3

A truck cask will carry from one to three PWR elements or from two to seven BWR elements. Such a cask will be cylindrical in shape, approximately 1.5 m in diameter and 5 m long, and will will weigh up to 35 MT. A rail cask will carry up to 7 PWR elements or 18 BWR elements. The rail casks, also cylindrical in shape, will be only a little larger than truck casks but may weigh 70 to 100 MT (see Figures 4, 5, and 6). Neutron absorbers in multiple element casks may be necessary to assure nuclear criticality safety.

Radiation shielding is provided in the cask walls. Thick steel, lead, or uranium, which accounts for most of the cask weight, is used to attenuate gamma radiation from the fission products. Hydrogeneous material such as wood or water is used to absorb the neutron radiation from the spontaneous fission and alpha-neutron reactions with oxygen in the fuel due to Cm-242 and Cm-244 present in significant quantities in fuel which has been irradiated to more than about 20,000 MWD/MT.

The cask also must provide the means to dissipate the heat produced by radioactive decay. Water is usually used in the central cavity as a heat transfer medium or primary coolant to transfer the decay heat from the fuel elements to the body of the cask. The heat is usually dissipated to the air through fins on the surface of the cask container by natural processes. For some of the larger casks, air is forced over the fins by blowers to increase the cooling. In one design, heat exchangers using a secondary coolant with cooling coils running into the body of the cask literally pump the heat out and into the atmosphere; the primary coolant is not brought outside of the cask cavity. Reliable redundant systems are used where mechanical systems are relied on to assure cooling for safety.

Spent fuel shipping casks are designed to withstand severe transportation accidents without significant loss of contents or increase in external radiation levels. The casks are protected from the damaging effects of impact, puncture, and fire by thick outer plates, protective crash frames, or other protective overpacks, or are otherwise designed to control damage. The cavity is usually protected from excessive pressure by a rupture disk or a pressure relief valve.

C. Transport Conditions

At present, all shipments of irradiated fuel are made exclusive use, by truck or rail. Some barge shipments may be made in the future. It is unlikely that such shipments will be shipped in general freight as less-than-truckload or less-than-carload lots.

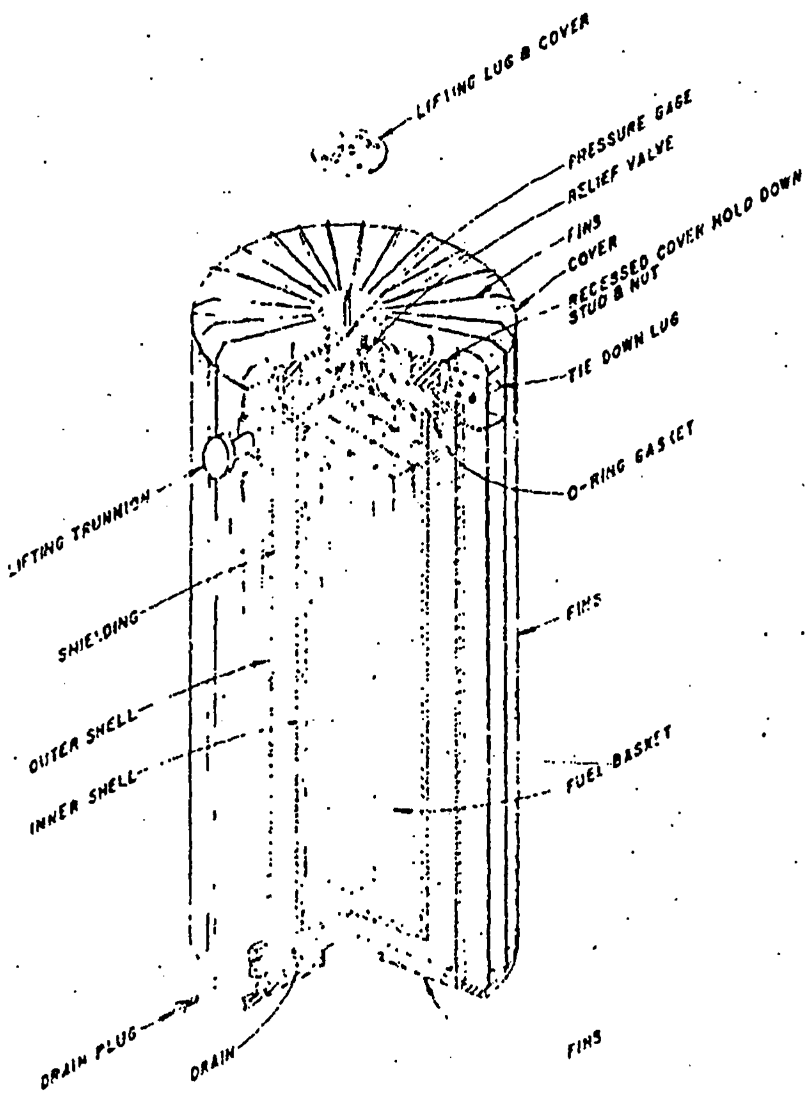

FIGURE 4 Cutaway Diagram of a Shipping Cask Showing the Principal Components.

Reproduced from best available copy.

Approximate:

Length	5.3 meters
Diameter	1.6 meter
Weight	
empty	55 tons
loaded	67 tons

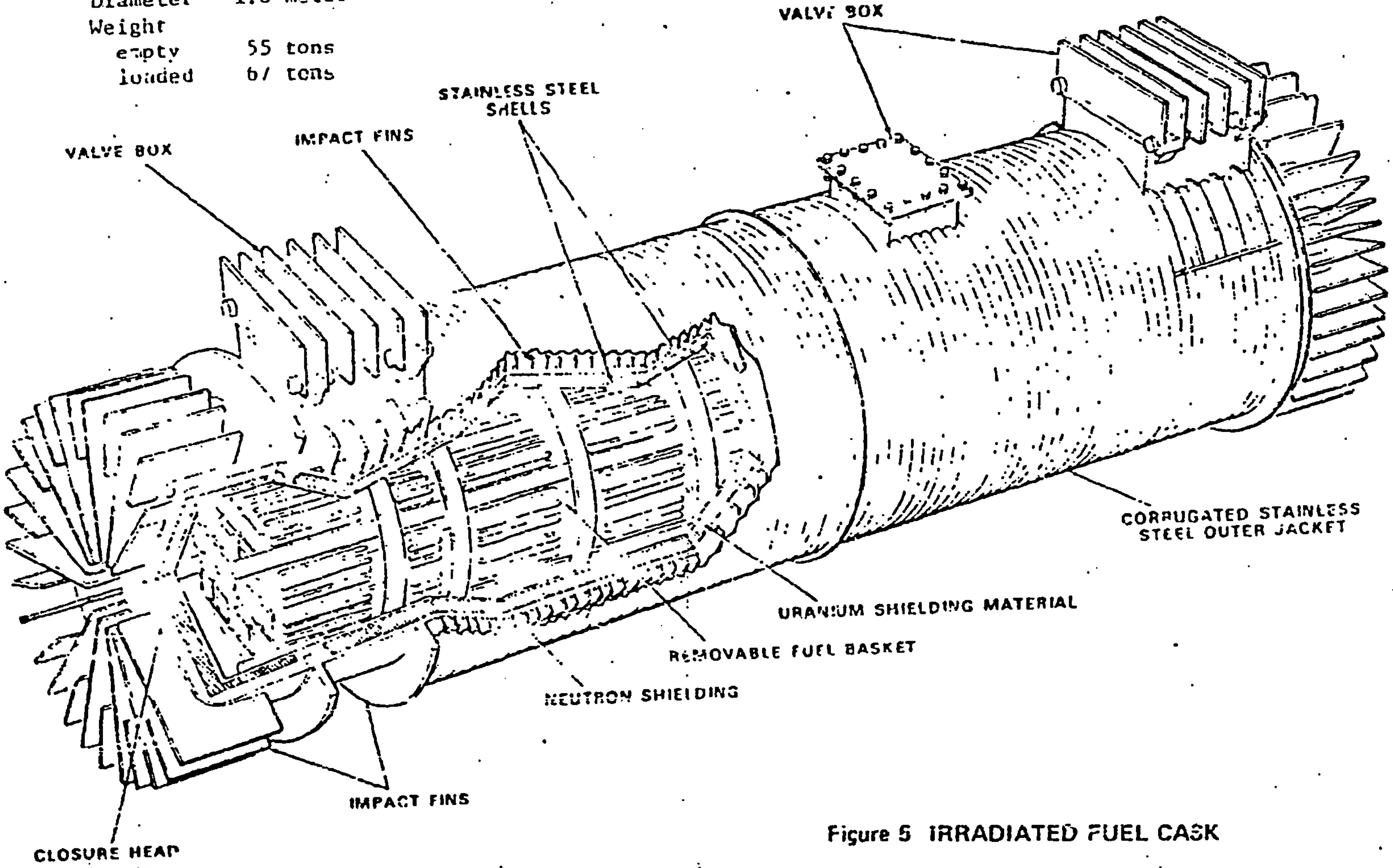

Figure 5 IRRADIATED FUEL CASK

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Approximate weight of cask
and shipping assembly:

empty 70 tons
loaded 82 tons

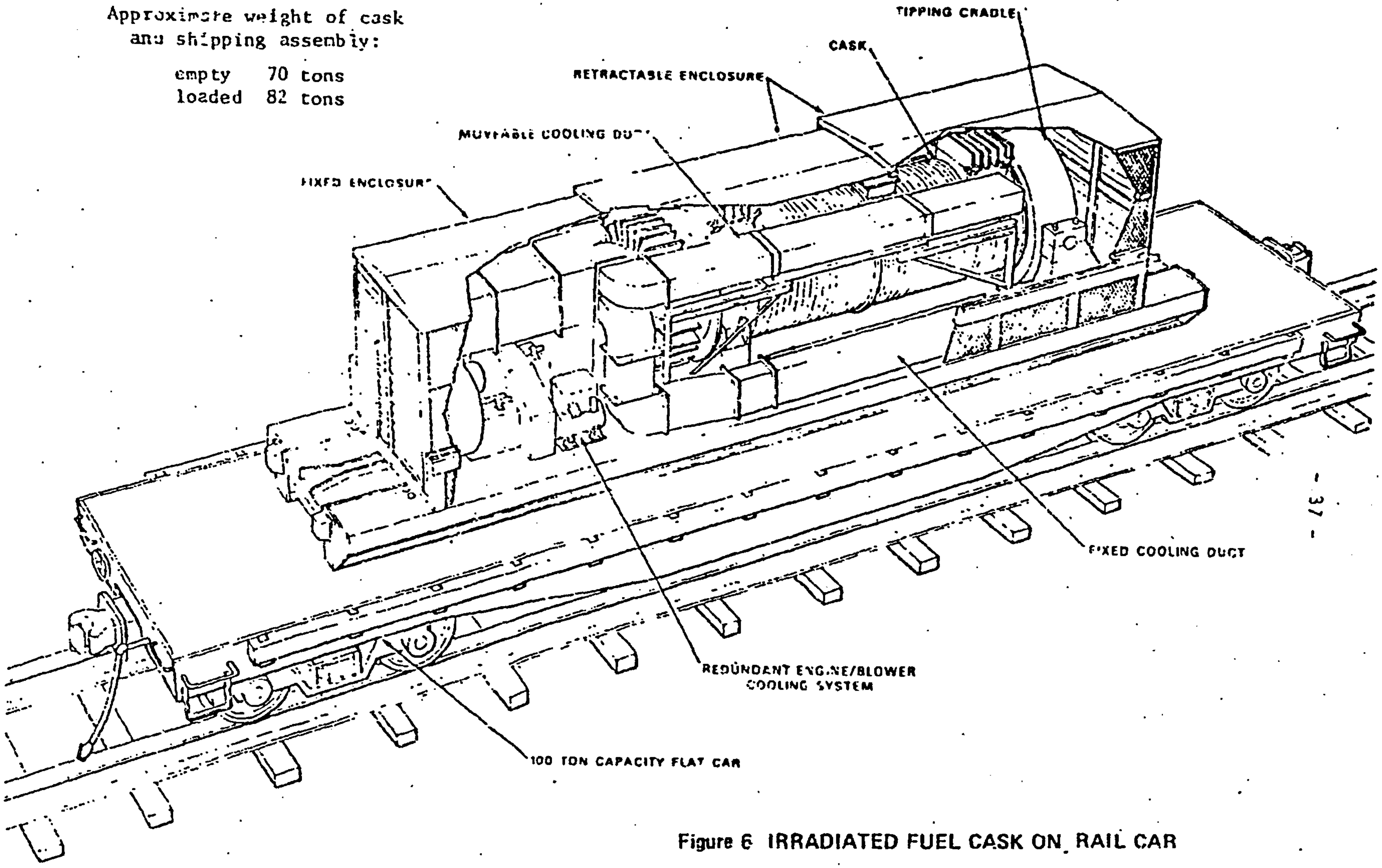

Figure 6 IRRADIATED FUEL CASK ON RAIL CAR

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The estimated average distance from the nuclear power plant site to the fuel reprocessing plant over which the irradiated fuel would be transported is 1000 miles. This journey would require an average transit time of about 3 days by truck and about 8 days by rail. Barge shipments might require 10 or 15 days, depending on the route.

Shipments by motor freight of spent fuel may be made from all reactor sites to all reprocessing sites. Many nuclear power plants do not have rail service directly onto the plant site. For this reason, those plants are either restricted to highway shipments using the lighter weight casks or must rely upon intermediate trucking by special equipment to the nearest railhead. Only a few of the nuclear power facilities are located on navigable waterways. Also, only one fuel reprocessing plant currently operating or planned will have the capacity of receiving shipments by water.

If barge transport of casks is to be used, construction of docking facilities might be required, at a cost of from $25,000 to $1,000,000.[17] Because of the probable high cost, docking facilities are unlikely to be built only for the purpose of shipping irradiated fuel elements. If docks are required for other purposes, they may be used for the transportation of irradiated fuel.

There are no plans at present to ship irradiated fuel elements by air. The possibility of air shipment is under study in the airlines industry to determine if the economics and safety aspects are acceptable. In all cases, air shipments will require truck movement from the nuclear power plant site to the airport and from the airport to the fuel reprocessing plant site.

D. Effects on the Environment

Normal Conditions

1. Heat

The rate of release of heat to the air from each cask will be about 10 to 70 kilowatts or from about 35,000 to 250,000 Btu/hr, depending on the type and amount of irradiated fuel contained. This might be compared to the rate at which waste heat is released from a 100 horsepower truck engine operating at full power, which is about 50 kilowatts or 180,000 Btu/hr. The temperature of the air which contacts the loaded cask will be increased a few degrees but the temperature of the air a few feet from the cask would remain unaffected. The longest period of time that one would expect the loaded cask to be present at a particular location, other than at the nuclear power plant site or the fuel reprocessing plant site, would be about eight hours, e.g., during the driver rest periods at truck stops or terminals and in rail yards awaiting make-up of trains. Because the amount of heat is small and is being released over the entire transportation route, no appreciable effect on the environment will result.

The DOT regulations[9] limit the temperature at any accessible surface of the cask to not more than 180°F at any time during transport, including stopover points. Although this temperature is not high enough to present a fire hazard, it could cause burns if contacted by bare skin. Since access to the cask is controlled to a large extent and each package is labeled with a "Radioactive" warning label, the likelihood of people becoming burned in this manner is quite low. There have been no known cases of such burns from those shipments which have been made with surface temperatures near the 180°F limit.

2. Weight and Traffic Density

Shipping the irradiated fuel from a single refueling of the reactor to the fuel recovery plant will require an average of 60 truck shipments, 10 rail car shipments, or 5 barge shipments. The casks are returned empty to the reactor. The weight of the spent fuel in a loaded cask constitutes only 2 or 3% of the total weight of the loaded cask. Because the cask being returned empty weighs almost as much as the cask loaded with irradiated fuel, the weight and number of shipments of empty casks must be considered in assessing the impact on the environment of the shipment of irradiated fuel. Therefore, considering return shipments of the cask, shipping the irradiated fuel will involve a total of 120 truck movements, 20 rail car movements, or 10 barge movements each year.

The total number of such shipments is too small to have any measurable effect on the environment due to the resultant increase in traffic density.

State highway weight restrictions limit the gross weight of trucks for routine shipments so that the gross weight of casks is limited to about 25 tons. Shipments of casks weighing up to about 35 tons may be allowed in most States under a special overweight permit. The States often prescribe special routing for overweight shipments and in some cases restrict the period during which the truck can travel. Repetitive shipments of overweight loads may cause breakup of the roadway. Some irradiated fuel shipping casks may require overweight permits. The number of such shipments is limited to about 60 round trips per year per reactor. That number of overweight shipments would not be expected to have any adverse effect on the roadways. Rail shipments of 50 to 100 tons of other commodities, such as coal, are routinely handled so the rail shipment of a 70-ton cask would offer no unusual loadings of rail facilities. Barges routinely transport cargoes weighing more than 100 tons.

3. Radiation

a. Regulatory Limitations

The radiation level at the surface of packages of radioactive material is limited by the DOT regulations[6] to no more than 200 mrem/hr, and at 3 feet from the surface to no more than 10 mrem/hr. If the shipment is made in a closed truck or rail car, the radiation level at 3 feet from the surface of the cask may be as high as 1,000 mrem/hr, provided that the radiation level does not exceed 200 mrem/hr at the surface of the vehicle, 10 mrem/hr at 6 feet from the surface of the vehicle, and 2 mrem/hr in either the driver's compartment or in a normally occupied position in a rail car.

Because of the large size of the packages used for shipping irradiated fuel, the limiting factor will be the radiation level at either 3 feet from the surface of the package, or six feet from the vehicle. Therefore, the radiation levels at the package surface will be considerably below those allowed by the regulation.

Based on actual experience, radiation levels around some irradiated fuel casks may exceed 200 mrem/hr at the surface of the cask, but will meet the limitations of 1,000 mrem/hr for closed vehicle shipments. In order to meet the limitation of 10 mrem/hr at 6 feet from the vehicle surface, the level will rarely exceed about 50 or 60 mrem/hr at the vehicle surface, or 25 mrem/hr at 3 feet from the truck or rail car.

Although a radiation level of 2 mrem/hr is permitted in a truck cab, the level based on actual experience is unlikely to exceed 0.2 mrem/hr, owing to the distance from the cask and shielding provided by intervening material.

b. Radiation Exposures

(1) Truck Drivers. Two truck drivers during a 1000 mile trip will probably spend no more than 20 hours in the cab

and about 1 hour outside the truck at an average distance of 3 feet from the cargo compartment. Under those conditions, each truck driver could receive about 30 mrem from an irradiated fuel shipment. Actual experience indicates that average exposures are much less than 30 mrem/trip; in most cases, less than 10 mrem/trip. The same driver is unlikely to be used for more than 30 shipments per year, in which case he would receive about 300 mrem in a year based on 10 mrem/trip. Based on 10 mrem/trip/driver, the cumulative annual dose to all drivers for 60 trips with 2 drivers on each trip would be 1.2 man-rem.

(2) Garagemen and Brakemen. For truck shipments, normal servicing of the truck will probably require two garagemen to spend about 10 minutes around the cab of the truck. Each could be exposed to about 0.02 millirem. The cumulative annual dose to all garagemen for 60 shipments would be about 0.002 man-rem.

For rail shipments, train brakemen would be expected to spend from 1 minute to 10 minutes each in the vicinity of the car during the trip, for an average exposure of about 0.5 mrem per shipment. With 10 different brakemen involved along the route, the cumulative dose for 10 shipments during the year is estimated to average about 0.05 man-rem.

(3) Freight Handlers. Irradiated fuel shipments are transported as full loads. Since the casks are not handled enroute, under normal conditions there would be no routine exposure of the carrier's freight handlers, either by truck or rail.

In-transit storage of these casks is unlikely except while mounted on the vehicle (truck or rail) at truck stopover points, in terminal yards, or in railroad switch-yards. There will be little, if any, across-the-dock handling of these casks outside of the nuclear power plant and the fuel recovery plant sites.

There is little likelihood that carrier personnel or members of the general public will get close to the side of the vehicle except in the case of transshipment, e.g., when the cask is transported by truck from the reactor to a nearby railhead and transferred from the truck to a railroad car.

All such handling must be done with cranes and heavy lifting equipment so that the exposure of persons occurs only during untying and tying down and hooking and unhooking lifting hooks. This might require 1/2 hour exposure at an average distance of 3 feet from the cask or about 100 mrem exposure for each of two persons handling the cask. If there were ten shipments to the railhead handled in this fashion, the cumulative annual dose would be about 2 man-rem. The crane operator and other workers in the area would be unlikely to receive any significant exposure.

In hauling the shipment to the railhead by truck, a distance of perhaps 20 miles, two truck drivers might spend an hour in the cab and perhaps 15 minutes outside the truck at an average distance of 3 feet from the cask. Assuming the radiation level in the cab is 2 mrem/hr and the level at 3 feet from the cask is 100 mrem/hr, each truck driver might receive as much as 30 mrem during each shipment. If the same two truck drivers were used for all ten shipments, each could receive as much as 300 mrem. The cumulative annual dose to all drivers would be about 0.6 man-rem.

(4) Barge Operators. A barge operator or tugboat operator who picks up the loaded barge at the nuclear power plant site will probably spend no more than an hour lashing the barge down, and checking lights and equipment at a distance of 50 feet from the cask, and perhaps a total of 10 minutes within 3 feet of the cask during the entire trip. His total dose would be about 4 mrem per trip. If two operators were involved, this would be a cumulative annual dose of about 0.04 man-rem for the five barge shipments.

(5) General Public--Onlookers. Members of the general public are normally excluded from loading and unloading operations, but some exposures might occur at enroute truck stops for fuel and eating and at railroad stations. Railroad cars carrying irradiated fuel shipments are placarded on both sides and trucks on both sides and the front and rear as "Radioactive." A member of the general public who spends 3 minutes at an average distance of 3 feet from the truck or railcar, might receive a dose of as much as 1.3 mrem. If ten persons, on the average, were so exposed during each shipment, the cumulative annual dose to such onlookers for the 60 shipments by the truck would be about 0.8 man-rem and for the 10 shipments by rail, about 0.1 man-rem.

Because of the conditions under which barges travel, onlookers are unlikely to be in a location where they would receive any significant exposure from barge shipments of irradiated fuel.

(6) General Public--along the route. Approximately 300,000 persons who reside along the 1000 mile route over which the irradiated fuel is transported might receive a cumulative dose of about 1 man-rem per year if the irradiated fuel is transported by truck and about 0.2 man-rem if transported by rail. An estimated 100,000 persons along the route might receive about 0.03 man-rem if transported by barge. In this case, the regulatory radiation level limit of 10 mr/hr at 6 feet from the vehicle was used to calculate the integrated dose to persons in an area between 100 feet and 1/2 mile on both sides of the shipping route. It was assumed the shipment would travel 200 miles per day and the population density would average 330 persons per square mile along the route, except that for barge, it is estimated that persons are within 1/2 mile of the barge route over only about 1/3 of that route. See Appendix D for the detailed calculations.

(7) Animals. The exposure of domestic animals or pets during transit is unlikely since the irradiated fuel is transported exclusive use.

(8) Film. Unexposed photographic film is not likely to receive any exposure during transit of irradiated fuel since, in most cases, there is no other freight loaded on the car or truck because of the weight and nature of the cask.

It is possible that a car or truck containing unexposed film could be parked adjacent to a truck or car containing irradiated fuel for several hours. The likelihood of this occurrence is so low that it is not practical to calculate it.

Accident Conditions

1. In-Plant Accidents

The "in-plant" radiological aspects of transportation of radioactive material are evaluated separately as part of the licensing procedures or contractual requirements and are not evaluated against the packaging standards and criteria for transportation. For that reason, the "in-plant" aspects have not been included in this analysis.

2. Off-Site Accidents

A cask must be so designed that in the unlikely event a shipment of irradiated fuel is involved in an accident, it is unlikely that there would be any release of radioactive material or increase in radiation levels outside of the package and under even the most severe accident conditions, releases above levels specified in the regulations are very unlikely. Although the consequences of a major release from an irradiated fuel cask could be severe, the low probability of such occurrences makes the risk from such accidents small.

a. Leakage of Coolant Under Other than Accident Conditions

The likelihood of leakage of coolant from a cask, under other than accident conditions, is very small because of the rugged, leaktight design of the cask and the procedures the shipper is required to follow to ensure leaktightness when preparing the cask for shipment.

The consequences of a leak depend on the amount of radioactive material which could be released by an undiscovered leak. A cask is required to be held at the origin until certain checks have been made including pressure, temperature, and checks for leakage. Any major leak would be discovered at the origin and corrected. If too much coolant were lost, it could cause overheating.

According to information supplied by the N-14 Committee of The American National Standards Institute, leakage of liquid at a rate of 0.001 cc/sec or about 80 drops an hour is about the smallest that can be detected by visual observation of a large container. It is expected that leakage at a rate exceeding 0.001 cc/sec would moisten a large enough area to be visible or would drip and probably would be detected and corrected at the reactor site. A leakage rate of .001 cc/sec on a large heated cask is expected to be evaporated as rapidly as it leaks out. Some fraction (perhaps 1%) of the radioactivity in the released liquid might be dispersed in the form of an aerosol. The exposure to people from such releases would be extremely small.

The AEC regulations limit[18] the contamination level in the coolant under normal conditions to 10^{-7} curies/cc of Group I (plutonium), 5×10^{-6} curies/cc for Group II (strontium and mixed fission products), and 3×10^{-4} curies/cc for Groups III and IV radionuclides (cesium and uranium). Based on 0.25% of the rods being perforated, we estimate about 1 μCi/cc of gross fission product activity might be in the cask coolant. Experience reported by Savannah River processing plant[19] indicates that the activity in water-filled casks ranges from 10^{-5} to 1 μCi/cc and that the activity is primarily cesium-137.

In 5 days, an undetected leak of 0.001 cc/sec would release 430 cc or about 400 μCi of activity. Under most conditions, that contamination would be retained on the surface of the cask and bed of the truck or railroad car.

b. Accident Conditions.

If transported by truck, it is estimated that a loaded cask would be involved in an accident about once in 20 reactor years and if transported by rail or barge, about once in 170 reactor years (see Appendix A).

Each cask is so designed and constructed that the probability is low of a cask being breached in the unlikely event it is involved in an accident. The form of the nuclear fuel is such that, should a breach occur, releases of radioactivity are unlikely and those releases that would occur are likely to be limited to gases and liquid coolant present in the cavity. The uranium, actinides, and most of the fission products would remain tightly bound in the oxide pellets. Some of the gases and most of the volatile and semivolatile actinides and fission products released from the oxide pellets would be retained by the cladding in the void spaces of the fuel rods.

The total amounts of the important gases, actinides, and gross fission products in low-enrichment fuel which has been cooled 150 days after irradiation at a power level of 30 MW/MT for a total of 33,000 MWD/MT are listed in Tables 4 and 5. The important activities in the void spaces of the fuel rods are shown in Table 7.

TABLE 7

FUEL ROD VOID SPACE ACTIVITY

Type of Radioactive Material	Total Inventory[16] 150 days cooling curies/metric ton	% in void spaces of fuel rods*	Activity in void spaces curies/metric ton
Kr-85	1.12×10^4	30	3.4×10^3
I-131	2.17	2	4.3×10^{-2}
Other fission products	4.38×10^6	0.01**	400
Actinides, (Pu, Am, Cm)	1.36×10^5	essentially none	neg.***
Xe-131m	3.27	2	0.1***
I-129	2×10^{-3}[20]	30	6×10^{-4}***
H-3	6.92×10^2	1	7***

* Realistic gap activities in terms of percent of total inventory prepared by AEC's Directorate of Licensing based on references 20 through 32.

** A conservative (high) value estimated on the basis of leaching the outer 1.2×10^{-5} inches from the surface of the uranium oxide fuel.

*** Due to the small amounts present, the dose contribution from Xe, I-129, H-3, and the actinides may be neglected compared to the doses from the other radionuclides.

The amount of radioactivity released relates to the number of fuel rods which are perforated. Penetration of the cladding would release some of the gases and gross fission products from the void spaces into the cask cavity and coolant. In the absence of a severe impact, it is believed conservative to assume that 0.25% of the fuel rods may be perforated. Even if all of the rods were ruptured, the radioactivity released would be unlikely to exceed 1.1×10^4 Ci of Kr-85, 0.1 Ci of I-131 and 1.3×10^3 Ci of other volatile and soluble fission products. Because of the cask design and quality control, the nature, form and physical properties of the fuel assemblies, the probability of such a release is so small as to be practically incredible.

c. Extended Fire.

Involvement of a cask in a fire lasting as long as 4 or more hours could cause loss of some neutron shielding and, if lead is used, loss of some gamma shielding. Releases of radioactive materials could be as much as those estimated above. The probability of an accident occurring in which such a long fire results is very small and the probability of a cask being involved in such an accident is so small as to be practically incredible.

d. Submersion in water.

If a cask is accidentally dropped into water during transport, it is unlikely to be adversely affected unless the water is deep. Most fuel is loaded into casks underwater, so immersion would have no immediate effects. The water would remove the heat so overheating would not occur. Each cask is required by § 71.32(b) to be designed to withstand an external pressure equal to the water pressure at a depth of 15 meters, and most designs will withstand external pressure much greater than that. If a cask were to collapse due to excessive pressure in deep water, only the small amount of radioactivity in the cask coolant and gases from perforated elements in the cask cavity are likely to be released. The direct radiation would be shielded by the water. About 10 meters of water, which is the depth of most storage pools, would be ample shielding for radiation from exposed fuel elements.

From our evaluation, the sinking of a cask in deep water would not result in serious radiological consequences. The most likely mechanism for loss of containment from external water pressure would be through failure of the pressure relief valves. This would result in an inflow of water and subsequent release of some of the contaminated coolant and radioactive gases present in the cask cavity. If all of the coolant and gases were released, the total activity might be on the order of 300 curies, most of which would be krypton-85 gas. The vast quantities of water available at the depth at which such a failure might occur would provide sufficient dilution so that it is unlikely there would be any significant radiation exposure or environmental impact.

The fuel elements, which contain most of the radioactive material, provide excellent containment. In an operating reactor, the fuel elements are under water at elevated temperatures and pressures on the order of 1000 to 2000 pounds per square inch guage. Thus exposure to water pressures at depths of 600 to 1200 meters should have no substantial effect on the fuel elements themselves.

Except under very unusual circumstances in which the cask could not be located or was submerged in extreme depths, the cask probably could be recovered with normal salvage equipment. If the cask and elements were not recovered, there would be a gradual release of radioactive material over a long period of time, several hundred years. Considering the extremely low probability of occurrence, the major reduction in radioactivity due to radioactive decay, and the dilution that would be available, there would be no significant environmental impact from this gradual diffusion of the radioactive fuel.

Accident Risk

Considering the low probability of a shipment of irradiated fuel being involved in an accident, the requirements for package design and quality assurance, the nature and form of the irradiated fuel, and the controls exercised over the shipment during transport, it is concluded that the radiation risk to the environment from irradiated fuel in transportation accidents is small.

VI. DETAILED ANALYSIS OF THE ENVIRONMENTAL IMPACT OF TRANSPORTING SOLID RADIOACTIVE WASTE FROM A TYPICAL LIGHT-WATER NUCLEAR REACTOR IN ACCORDANCE WITH PRESENT REGULATORY STANDARDS AND REQUIREMENTS

A. Characteristics

Solid waste, primarily sludges and resins, is estimated to amount to about 3,800 cubic feet per year from a BWR (see Table 8). Of this amount, 120 cubic feet of cleanup sludge per year will contain 13 curies per cubic foot and require Type B packaging. The remaining 3,680 cubic feet will contain about 0.09 curies per cubic foot and can be shipped as low specific activity materials, or in Type A packaging. Solid waste from a PWR is similar in form but the total is about 1,000 cubic feet per year, about 24% of which will be resins containing 0.6 curies per cubic foot, 75% sludges containing 0.01 curies per cubic foot, and 1% resins and sludges containing up to 15 curies per cubic foot.

TABLE 8

SOLID WASTE FROM 1100 MWe BWR

	Volume[33] (ft^3/yr)	Radio-[33] activity (Ci/ft^3)	No. of* Drums @ 7.2 ft^3/drum	Ci/drum
Cleanup Sludge	120	13	67	23.3
Condensate sludge	2100	0.14	1166	0.25
Waste sludge	920	0.01	511	.02
Waste bead resin	60	0.01	34	.02
Cond. bead resin	600	0.06	334	0.01
Totals	3800 ft^3/yr		2112	

* Assuming the waste is mixed with concrete in the ratio of 1.6 ft^3 of waste and 5.4 ft^3 of concrete per 7.2 ft^3 (55-gallon) drum.

In addition, soft solid wastes such as contaminated clothing, rags, paper, gloves, and shoe coverings containing low levels of contamination will be generated. This low level waste, probably compacted to reduce the volume, may be shipped in 55-gallon drums. Each year, on the average, one might expect 30 to 50 drums (one truckload or a part of one carload), each drum containing 500 pounds of compacted material contaminated with 0.5 curies of corrosion, activation, and fission products, to be shipped for disposal.

B. Packaging

Under the regulations of the DOT, solid wastes may be shipped in strong industrial, Type A, or Type B packages depending on the amount of radioactivity. Typically, the waste is compacted, or solidified in a mixture of vermiculite and cement in steel drums. The drums when filled weigh from 500 to 800 pounds with an average weight of 700 pounds. The drums are normally made of 18-gauge steel with 16-gauge "clamp-on" lids. Wastes which are low specific activity materials, or Type A quantities in drums, may be shipped without further packaging. Type B quantities must be shipped in Type B packages; these might be drums in an "overpack" (i.e., a protective outer container) which provides impact and thermal protection for the drum or shielded flasks designed to meet Type B requirements.

C. Transport Conditions

About 2,100 drums weighing an average of 700 pounds each would be required to ship the solid waste to the burial grounds. The waste is shipped either by truck with 40 to 50 drums per truckload, or by rail with 200 to 250 drums per car. This will involve 46 truckloads or 11 carloads per year. Barge and air shipments are unlikely.

All shipments will likely be made under exclusive use--full load--arrangements. Such shipments will be transported an average distance of 500 miles (a minimum distance of 50 miles to a maximum of 3,000 miles). The average transit time will be about 3 days by truck and 7 days by rail.

D. Effects on the Environment

Normal Conditions

1. Heat. Most of the packages of waste would have no readily detectable heat output. Those containing the cleanup sludges might generate about 0.1 watt or 0.4 Btu/hr of heat per package which is negligible as far as effect on the environment is concerned.

2. Weight and Traffic Density. The number of shipments per year, about 46 by truck or 11 by rail, is too small to have any measurable effect on the environment due to the resultant increase in traffic density.

The number of drums of waste per vehicle can be adjusted so that the truck can stay within the weight restrictions of the State (usually about 25 tons) or the rail car can meet the railroad limitations on gross car weight. There should be no need for overweight permits, and therefore no excessive loads on the roadbeds or bridges for major routes.

3. <u>Radiation</u>

a. <u>Regulatory Limitations</u>. Drums of wastes must meet the regulatory limitations on external radiation levels described in the previous section on Regulatory Standards. In practice, most of the drums will contain such small quantities of radioactivity that the radiation levels at the surface of the drums will be less than 200 mr/hr.[33] Radiation levels at the edge of the load, which is the surface of the truck or rail car, are unlikely to exceed 50 to 60 mrem/hr; at 3 feet from the surface of the vehicle, 25 mrem/hr; and at 6 feet from the surface of the vehicle, 10 mrem/hr. The radiation level in the truck cab is not likely to exceed 0.2 mrem/hr.

b. <u>Radiation Exposures</u>

(1) <u>Truck Drivers</u>. Two truck drivers during a 500 mile trip would probably spend no more than 20 hours in the cab and about 1 hour outside the truck at an average distance of 3 feet from the cargo. Under those conditions, each truck driver could receive about 30 mrem from a solid waste shipment. Actual experience indicates that average exposures are much less than 30 mrem; in most cases, less than 10 mrem/trip. The same driver is unlikely to be used for more than 30 trips each year, in which case he would receive about 300 mrem in a year based on 10 mrem/trip. The cumulative annual dose to all drivers might be about 1 man-rem.

Discussions with companies who ship or carry packages almost daily revealed[34] that the exposures of drivers and handlers who were routinely monitored were very low, many showing no exposure above background, even when such persons are assigned the regular job of transporting radioactive materials.

(2) Freight Handlers. Shipments of wastes are transported as "full-loads." Since the drums are not handled enroute, there would be no exposure of the carrier's freight handlers, either by truck or rail.

(3) Garagemen. For truck shipments, normal servicing of the truck would probably require two garagemen to spend no more than 10 minutes around the cab of the truck. Each would be exposed to about 0.02 millirem. The cumulative annual dose to garagemen for 46 shipments would be about 0.002 man-rem.

(4) Brakemen. For rail shipments, train brakemen would probably spend from 1 minute to 10 minutes each in the vicinity of the carload of drums of waste during the trip, for an average exposure of about 0.5 mrem per shipment. If 10 different brakemen were involved along the route, the cumulative annual dose would be about 0.05 man-rem for the 11 shipments.

(5) General Public -- Onlookers. Members of the general public might be exposed to radiation from shipments of waste at enroute truck stops for fuel or eating or at railroad stations. Car loads of solid waste shipments will be placarded on both sides and truckloads on both sides and the front and rear as "Radioactive." A member of the general public who spends 3 minutes at an average distance of 3 feet from a loaded truck or car might receive a dose of as much as 1.3 mrem. If 10 people were so exposed during a shipment the cumulative annual dose to such onlookers for the 46 shipments by truck would be about 0.6 man-rem, and for the 11 shipments by rail, about 0.1 man-rem.

(6) General Public -- Along the Shipping Route. An estimated 150,000 persons who reside along the 500 mile route over which the solid waste is transported might receive a cumulative dose of about 0.4 man-rem per year if the waste were transported by truck and about 0.1 man-rem if transported by rail. These doses were calculated for persons in an area between 100 feet and 1/2 mile on either side of the shipping route, assuming 330 persons per square mile, 10 mr/hr at 6 feet from the vehicle and each shipment traveling 200 miles per day. See Appendix D for the detailed calculations.

(7) Animals. The exposure of domestic animals or pets during transit is unlikely since the waste is transported as a "full-load."

(8) Film. Unexposed film is unlikely to be loaded on the same vehicle as a load of waste, and hence is unlikely to receive any radiation exposure. It is possible that a car or truck containing film could be parked adjacent to the carload or truckload of waste for several hours. The likelihood of this occurrence is so low that it is not practical to calculate it.

Accident Conditions

1. In-Plant Accidents

The "in-plant" radiological aspects of transportation of radioactive material are evaluated separately as part of the licensing procedures or contractual requirements and are not evaluated against the packaging standards and criteria for transportation. For that reason, the "in-plant" aspects have not been included in this analysis.

2. Off-Site Accidents

The likelihood of leakage of radioactive material from a package of solid waste is small because of the solid form of the material and the leaktight design of the containers. Both the solid form of the material and the small amount of radioactivity per unit mass limit the adverse effect in the unlikely event a release should occur.

a. Improperly Closed Packages

In the shipment of a large number of packages of solid wastes, it is possible that some of the drums or packages may not be properly closed as a result of human error. It is estimated that about one in 10,000 packages may not be properly closed when shipped. In the unlikely event that an improperly closed package comes open, the solid form of the material, either as compacted soft wastes or consolidated solid wastes, makes it highly unlikely that other than a small release of radioactivity will take place. No significant radiation exposures would be likely to result. However, cleanup costs might amount to a few thousand dollars.

b. Accident Conditions

A truckload of solid waste may be involved in an accident about once in 25 reactor years and a rail carload about once in 250 reactor years (see Appendix A).

The packages used for the waste are so designed and constructed and the solid form in which the waste is shipped is such that, in the unlikely event a shipment of solid waste is involved in an accident, it is unlikely that the radioactive material would be released. Based on the results of an instrumented test[35] in which a semitrailer truck loaded with drums was crashed into an immovable barrier at 42 miles per hour, it is highly unlikely that more than 50% of the Type A packages or any of the Type B packages would be damaged in an accident. Most of the radioactivity is tightly bound in the waste and most of the waste is in a massive, solid form. Unless fire ensues, the amount of radioactivity which becomes airborne in the unlikely event a drum or package were to be broken open is unlikely to exceed a very small fraction (less than 0.1%) of the activity of the contents of that drum. In a fire, combustible wastes may be burned but most of the radioactivity in waste burned in a fire will remain in the ashes.

Soft solid wastes such as paper, contaminated clothing, etc., compacted and placed in drums are typical Type A packages of solid waste. Each may contain as much as 1 curie of activation and fission products, primarily Fe-59 and Cs-137 distributed throughout about 500 pounds of waste.

In the case of the consolidated solid wastes, e.g., concreted resins, sludges, etc., as much as 100 curies of activation and fission products may be contained in 700 pounds of waste and concrete in a 55-gallon drum or in concrete- or metal-shielded flasks. Because of the form of the waste, it is extremely unlikely that the contents would be released in any accident.

The amounts of radioactivity contained in each drum of waste are small in most cases. Based on the data presented earlier, about 3 percent of the drums of waste would contain compacted wastes with very low levels (millicurie amounts) of contamination and 95 percent would contain solid wastes with a total radioactive content averaging less than 0.3 curie per drum. In the unlikely event such packages are broken open in an accident, the consequences of a release would be very limited. Only about 2 percent of the drums or packages of waste would contain curie quantities of radioactive material and they would be required to be Type B packages, i.e., so designed that they would be unlikely to release their contents in an accident. Even if the entire contents were released, the solid form would limit the amount dispersed to a small fraction of the total activity.

Accident Risk

The probability of a shipment of solid radioactive waste being involved in an accident is very small. Because of the package design, quality assurance, and nature and form of the waste, a release is unlikely in an accident. In the event a release occurs, the small amount of radioactivity in most packages of waste and the fact that the radioactive material is tightly bound in a massive solid makes it highly unlikely that any serious radiation exposures would occur. Therefore, the radiation risk to the environment from solid radioactive waste in transportation accidents is small.

VII. POSSIBLE ALTERNATIVES AND ADDITIONS TO THE TRANSPORTATION METHODS ANALYZED

Under normal conditions there are no effects on the environment which would be considered adverse, and although the consequences of credible accidents are serious, the probability is so small that the overall risk is not sufficient to justify any significant effort to further reduce the consequences.

The following alternatives and actions were examined:

A. Routing

The probable routing of shipments of unirradiated and irradiated nuclear fuel and solid radwastes is indicated in some Environmental Reports for individual nuclear power plants. It is not intended that the shipments be restricted to these routes since the safety standards of the AEC and DOT do not rely on restriction of routing for assuring safety in transport.

The regulations of the States impose controls on weights of loads on roadways and bridges. Also, in some cases municipalities and bridge, tunnel, and turnpike authorities place restrictions on travel at specific periods of the day or night and over certain sections of routes. These latter limitations may affect the choice of routes.

Routes for shipping radioactive material could be required to be selected so as to avoid centers of population, special risk areas due to local road or rail conditions, areas of high accident frequency, extremes in ambient conditions such as very cold or very hot weather, high elevations, and delays. Such restrictions could reduce the probability of an accident occurring in many cases. However, if the shipping distances were increased to avoid the conditions, the accident frequency could be increased. Examination of local conditions would be required in each case to determine whether such restrictions would be advantageous or not.

Requiring radioactive material shipments to be shipped over routes which avoid centers of population would reduce the radiological consequences of those accidents in which a release of radioactivity is involved or direct radiation exposure of persons in the area results. This follows, since the dose would be smaller if the number of people in the affected area were smaller. The risk from accidents, however, involves both frequency and consequences. If the number of miles traveled is increased by the special routing restriction, the frequency of accidents will be increased unless the probability of an accident is smaller for the "special route," since the number of accidents is proportional to the number of miles traveled. Also, the risk from accidents due to common causes far overshadows the risk due to radiological effects. In truck accidents, for instance, non-fatal injuries occur in 33% of all truck accidents and fatal injuries in 3% of all truck accidents,[36] whereas the radiological effects occur in only a very small fraction of all accidents. Experience[37,38] and the statistics analyzed in this report show the probability of an accident occurring which causes any radiological effects is extremely small. Special routing to avoid centers of population to reduce the radiological effects, which are already small, can be expected to have only a very small effect. Therefore, any reduction in the already very small risk from radiological effects may be outweighed by an increase in the risk from common causes.

At present, truckers carrying hazardous goods are required by DOT regulations[39] to avoid congested places insofar as is practicable. Truck routes usually are chosen to move traffic along and for that reason usually avoid congested areas. Carriers use Interstate highways whenever possible. Interstate highways avoid centers of population in most cases. Although the use of divided highways and routes around population centers may reduce the probability of an accident occurring per mile, the severity of those accidents which do occur will be increased because of the higher rate of speed of the vehicle.

There are no specific regulatory requirements with regard to routing of hazardous materials shipments by rail. Severe rail accidents involve high speeds and frequently occur because of faulty roadbeds or equipment. Roadbeds connecting centers of population are used more frequently than off-the-main-line roadbeds and generally are better maintained for that reason. Further, accidents occurring inside city limits are unlikely to be as severe as those outside the city limits since speeds are restricted somewhat, and emergency equipment is more readily available. For these reasons, it appears likely that for rail shipment, the frequency of severe accidents may be greater for shipments made on routes chosen to avoid centers of population than if those same shipments were made on "main line routes" between population centers.

B. Escorts

Escorts, in separate vehicles or cars, could be required to accompany the shipments. They could be equipped to monitor the area and take corrective action in case of an accident. Escorts who survive could assist in control of any accident, but probably could not reduce the effects of immediate releases such as releases of noble gases and iodine. It does not appear likely that a requirement that escorts accompany a shipment can be justified in view of the low probability of a severe accident occurring in which an escort would be effective.

To be effective, escorts would have to be provided for each major shipment of radioactive material. Although an escort in a separate vehicle might mitigate the consequences of some accidents and reduce the already small probability of the shipment vehicle being involved in an accident, the escort vehicle has a probability of being involved in an accident at least equal to that of the shipment vehicle. Because injuries occur in 13% of all motor vehicle accidents, the increased number of injuries due to accidents involving the escort vehicle outweighs the small probability that escorts could reduce the consequences of the severe accident, less than 0.5% of all accidents.

C. Longer Storage of Spent Fuel

The amount of radioactivity and decay heat in the irradiated fuel can be reduced by holding the irradiated fuel in the storage pool at the reactor for long periods of time.

For purposes of shipment, the radioactive decay that takes place in irradiated fuel during the first 90 days after removal from the reactor is considered important. During that time most of the iodines decay to small values, the noble gases are reduced, and other short-lived radionuclides decay so that the amount of heat generated is greatly reduced. The difference in radioactivity inventory and decay heat between 90 days and 150 days is not considered to be significant for shipment. Therefore, shipment anytime after 90 days of cooling time is considered to be within the scope of this analysis. Shipment in less than 90 days cooling time would require reexamination of the added risk and potential benefit.

By storing the fuel for a full year instead of 150 days, the radioactivity and decay heat could be reduced by a factor of 2, and storage for 10 years would reduce them by a factor of 10. Storage beyond 150 days gains little in terms of reducing the inventory compared to the required increase in storage capacity for the nuclear power plant, fuel inventory costs, and the additional precautions necessary to assure that the risk is not greater because of the extra fuel on hand. On balance, it does not appear storage beyond 150 days is warranted.

D. Lower Radiation Levels Outside of Packages

It is possible to design and build heavier packaging with additional shielding or, by reducing the amount of radioactive material in a package, to reduce the radiation levels outside of the package. Additional shielding for most container designs would be added to the outside of the present shielding to avoid reducing the capacity of the container. The fractional increase in the weight of the container due to the added shielding would be more than the fractional increase in shielding thickness. The costs increase as the ratio of weight of container to weight of the contents increases. Additional shielding also increases the initial cost of the container.

The weight of present designs of casks is approaching the limits of the available handling and transport facilities. Extra package weight means a smaller number of packages per vehicle, which would mean more shipments. More shipments would be required if the content of present packages were reduced. Increasing the number of shipments increases the frequency of accidents and thereby increases the impact on the environment.

Taking into account the costs associated with additional shielding, weight limitations of available facilities and equipment, and the present state of the technology, the Staff concludes that the radiation levels associated with present designs of casks are as low as practicable.

E. More Stringent Accident Damage Test Criteria

The radiological risk due to accidents involving packages of radioactive material might be reduced by imposing more stringent accident damage test criteria on package designs.

Experience and estimated probabilities and consequences of accidents indicate the radiological risk in transport accidents which result from packages which meet the present accident damage test criteria is small (see Appendix B). Increasing the severity of the test conditions would require heavier or larger packaging designs to meet the criteria. Extra weight of packaging would reduce the ratio of weight of radioactive contents to package weight. Larger and heavier packages, in most cases, would mean a smaller number of packages per vehicle. The reduced ratio and fewer packages per vehicle would increase the number of shipments required to be shipped from an individual reactor. Increasing the number of shipments would increase the number of accidents in which such shipments would be involved.

Because the radiological risk is so small, imposing more stringent test criteria can achieve only a relatively small reduction in that risk. An increase in the number of accidents in which shipments of radioactive materials are involved tends to offset that advantage, because the overall risk from both radiological and common (i.e., non-radiological) causes is proportional to the number of accidents and the risk from common causes, although small (see Appendix C), is greater per accident than the risk from radiological causes.

Changes in the accident damage test criteria for radiological safety do not appear to be warranted in view of the small radiological risk as evaluated in this report. Considering the small overall risk in accidents and the present balance of radiological vs. common cause control, we conclude that the present accident damage test criteria provide control over the radiological risk to a level as low as practicable.

F. Nuclear Parks

The term "nuclear park" applies to a nuclear industry complex or cluster in which the nuclear fuel is fabricated, used, and reprocessed on the same or contiguous sites. This requires that fuel fabrication and fuel recovery facilities be located in the cluster with the nuclear power plant. In such a cluster, transportation of unirradiated and irradiated nuclear fuel for the power plant would be limited to movement on the site.

When and if nuclear parks are developed, this will minimize the risk from transportation of nuclear fuel.

APPENDIX A

ANALYSIS OF TRANSPORTATION ACCIDENTS

Introduction

One of the purposes of regulations applicable to the transportation of radioactive material is to assure that the risk of injury or damage to property from accidents in transport is low. With respect to radiological effects, this is achieved by a combination of limitations on contents, package design, and quality assurance requirements and controls exercised over storage and loading during transport. The probability of a vehicle carrying a shipment of radioactive material being involved in an accident in transport is not greater, and experience indicates it is less, than the probability of a vehicle of the same type transporting other goods being involved in an accident. In consideration of the environmental risks associated with transportation accidents, the probability of their occurrence and their consequences must both be taken into account.

As to the consequences, either the contents of each package must be limited so that in the unlikely event the contents were released, the consequences would not be serious or the package must be designed to prevent loss of contents or shielding and assure nuclear criticality safety under accident conditions. While the package design standards do not provide a completely indestructible package, it would require a very severe and highly unusual accident to breach a container. An analysis of the severity and probability of occurrence of accidents follows.

Experience

In the past 25 years, several millions of packages of radioactive material, including approximately 3600 packages of irradiated fuel, have been transported in routine commerce. The Department of Transportation estimates at present about 800,000 are shipped each year in the U. S. During that same period, there have been only about 300 accidents recorded[37,40] in which radioactive material were involved. None of these[4] resulted in serious injury of people as a result of the radioactive nature of the material. In only about 30% of those accidents was there any release of radioactive material from the package or increase in the radiation levels outside the package.

The accident statistics related above represent an excellent record of safety in the transportation of radioactive material. Since the accidents involving radioactive materials which have occurred are small in number and present only a limited range of conditions likely to arise in transportation and consequences of potential accidents, other data must be relied on for analysis and projection of the risk from accidents involving radioactive materials. One source of that data is accident experience with other hazardous materials. In 1971, 2255 accidents involving hazardous materials were reported to the DOT; only 10 involved radioactive materials. Two of the 10 accidents involved only empty radioactive material containers; 1 resulted in increased radiation levels; 2 produced low levels of contamination outside the vehicle and 1 (the Delta Airlines incident of December 31, 1971) produced contamination of a cargo compartment and some luggage and required a considerable amount of effort to clean up. In another accident, a shipment of UF_6 in large cylinders was involved in a train derailment but there was no release of UF_6. In 1972, through June 23, a total of 1696 accidents were reported to DOT; 8 involved radioactive materials. Only one of the 8 involved any release of radioactive material and that was a sealed source released from the package which was recovered with no residual contamination.

These statistics represent a distribution of accidents in transport skewed toward the severe end since the statistics include only reportable accidents, i.e., accidents which resulted in an injury or fatality or property damage in excess of $250.

Accident Model for Analysis

For analysis of data on accidents, an accident can be divided into a series of events and each event treated as a separate component. The progression of events involved in an accident which may result in damage from radiation effects are presented in a highly simplified model. Data for some of the events are available and for some are incomplete. Data on impact and fire accident probabilities and severities are available. There are considerable test data on resistance of radioactive material packages to impact and fire up to the level of the package design test criteria. Based on the data known about the stresses produced on packages in real transportation accidents, it is believed the present standards assure packages of radioactive material will withstand all but very severe, highly unlikely accidents. This is borne out by the statistics related above.

Some hazards are present in normal operation and some arise in normal operation. A threshold exists at each stage in the progression of events identified as an accident. If the hazard or combination of hazards at any stage fails to exceed the threshold, the process of the accident stops and no damage results. For example, if the impact energy absorbed by the vehicle is such that the energy transmitted to the package is below that which will cause failure of the package, and other forms of stress (such as failure of the tie-down or fire) do not develop, the progress of the accident stops at that point.

The packaging standards and criteria establish that threshold; for industrial type packaging and Type A packaging, the threshold is high under normal conditions and for Type B packaging, the threshold is high under both normal and accident conditions. The threshold of failure for packages is not known, although most Type A packages will withstand minor accidents and some will withstand severe accidents without loss of contents.[42] Type B packages are required to be designed to withstand specified accident damage test conditions; the point at which failure would occur is often not known. From an analysis of test results, it appears that some designs will withstand stresses well above the test conditions. Tests to destruction were made for certain types of containers in an attempt to better define that threshold.[43] The part of the package which fails and the type of failure, as well as the threshold of failure, vary from one type of package to another.

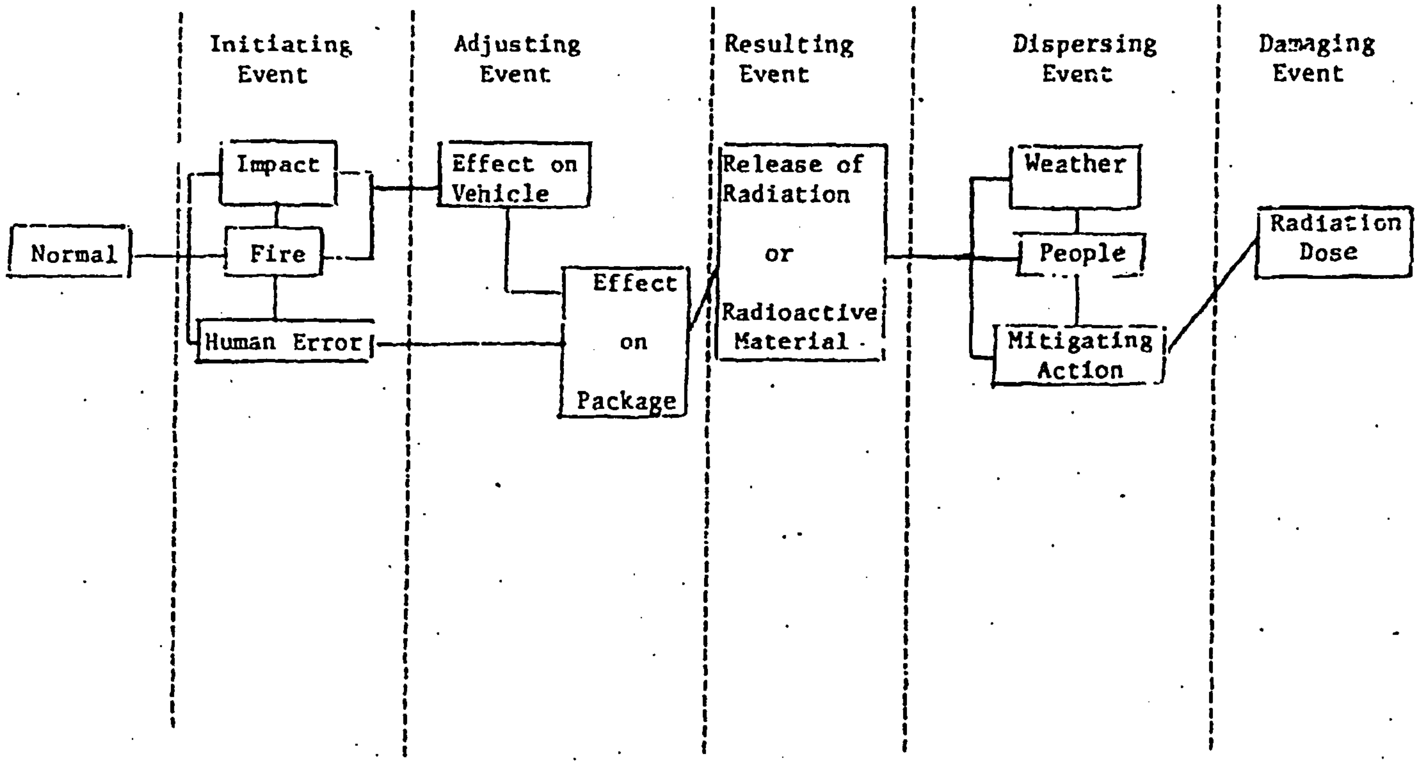
Transportation Accident Statistics[60]

The probabilities of accidents by truck, rail, and barge are derived below from statistics of accidents supplied by the U. S. Department of Transportation (DOT) for 1969 and 1970.[36,44,45] The conditions likely to be encountered in the accidents in terms of velocity of impact of the vehicle and incidence and duration of fire were developed from analyses made by Leimkuhler,[46] various statistics on frequency of fires, and information in the 1969 and 1970 accident statistics referred to above.

Accidents occur in a range of frequencies and severities. Most accidents occur at low vehicle speeds; the severity of accidents is greater at higher speeds but the frequency decreases as the severity increases. Accidents generally involve some combination of impact, puncture, and fire effects.

For purposes of this analysis, accidents are divided into five categories - minor, moderate, severe, extra severe and extreme.

FIGURE 7. TRANSPORTATION ACCIDENT RADIATION EFFECTS MODEL



Accident Statistics for Trucks

In 1969, large motor carriers[36] reported a total of 38,813 accidents involving death, injury, or property damage in excess of $250. The accidents included 19,682 injuries, 1,497 fatalities, with an overall accident rate of 2.46 accidents per million vehicle miles. For hazardous materials shipments, the accident rate was 1.69 per million vehicle miles. The overall accident rates per million vehicle miles for previous years are 3.2 for 1964; 2.3 for 1965; 2.4 for 1966; 2.4 for 1967; and 2.5 for 1968. Fifty percent of the reportable accidents involved collision with autos or buses, 15.5% collisions with other trucks, 14% collisions with fixed objects, 0.6% collisions with trains, 9.5% were roll-overs or run-offs, and 11.4% other types of accidents. Fire occurred in 1.57% of the reportable accidents.[47]

In truck accidents, severe damage to the package may be encountered in all types of accidents. Impacts which are likely to be most damaging are those on stationary, rigid objects, such as concrete abutments or bridge structures. In collisions with an object, yielding or crushing of the vehicle or the object with which the vehicle collides reduces the impact received by the package. Roll-overs usually occur at higher speeds, and must be considered as potential contributors to major damage of a package.

A study in 1960[46] showed the following percentages of accidents for the four ranges of truck speeds given. We have assumed those percentages apply to the four ranges of speeds used in our analysis of 0-30, 30-50, 50-70, and >70 mph.

TABLE 1

Type of Accident	Speed in MPH 0-32	32-52	52-72	>72
All accidents	23.7%	56.0%	19.8%	0.5%
Collisions with autos and buses	34%	42%	23%	1%
Collisions with other trucks	25%	72%	3%	0.1%
Overturns and other collisions	8%	69%	23%	0%

Truck fire data[36] indicate that fire is involved in about 0.8% of truck-truck collisions, 0.3% of the truck-auto collisions, 0.6% of truck-fixed object collisions, 2% of the truck-train collisions, and 1% of the roll-over/run-off accidents. Most fires involve only the fuel from the vehicle fuel tanks, and last less than 1/2 hour, unless other freight becomes involved. Only in the case of truck-truck collisions is there likely to be a larger supply of fuel involved, e.g., a collision with a gasoline tank truck or a truck loaded with paint. Some fires start from overheated tires or accidental ignition of cargo. Truck-auto, truck-bus, and single-vehicle accidents were considered to be essentially free of fires lasting longer than 1/2 hour.

It is assumed that only in truck-truck accidents is there a credible likelihood that fires would occur which last more than 1/2 hour, and then only when one of the trucks is carrying significant amounts of flammable liquids as cargo (e.g., tank trucks of gasoline or liquefied petroleum gas; or van trailers carrying barrels of paint). For lack of data on the percentage of trucks carrying flammables, it is conservatively assumed that at least one of the trucks in each truck-truck accident is carrying flammable cargo. Of all truck accidents, 15.5% involve other trucks, i.e., are truck-truck accidents having a potential for long fires.

Of the fires which do occur, it has been estimated[46] that 1% of the fires last more than one hour, 10% last between 1/2 hour and one hour and the balance, 89%, last less than 1/2 hour. Although there are fires in transport which last for several days, in most cases these involve the burning of only small amounts of fuel per unit time, and are of little consequence in terms of heat output.

The probabilities for truck accidents are listed in Table 3.

<u>Accident Statistics for Railroad Cars</u>

In 1969, for a total number of car miles of about 61 billion, the rail industry[44] reported a total of 8,543 accidents involving death, injury, or property damage in excess of $750, of which 4,971 were other than grade-crossing accidents. The accidents included 23,356 injuries, 2,29[illegible] fatalities.

In 1969,[44] the total number of accidents per million train miles was 9.89; for 1968, it was 9.16; and for 1967, it was 8.15. The average train length is about 70 cars.

The overall accident rate is 0.14 train accidents per million car miles. The accident rate for other than grade-crossing accidents is 0.08 train accidents per million car miles. Each accident involves an average of 10 rail cars, so the accident rate per car for other than grade-crossing accident would be about 0.8 car accidents per million car miles.

Twenty-one percent of the reportable accidents were collisions, 70% were derailments, and 9% were other types of accidents. About 1.5% of the rail accidents involved fire, most of them occurring in serious derailments in overland movements.

In rail accidents, severe damage to the cargo may be encountered in both collision and derailment type accidents. Rail grade-crossing accidents (train-truck or train-auto) rarely involve significant damage to cargo. Other collision type accidents which do not cause derailment are not likely to involve significant damage to a package. Accidents which have the highest probability of producing significant damage to shipment containers are overland derailment accidents which involve either impact of the packages on forward cars, or impact on the packages by rearward cars.

The accident rate of 0.8 car accidents per million car miles for other than grade-crossing accidents was used as the probability of a railroad car carrying a shipment being involved in an accident that might cause damage to that shipment. The overall accident rate of 0.14 train accidents per million car miles was used in estimating the effects from common causes of a car being involved in an accident.

An unpublished study by the DOT of the total accidents that occur at various speeds indicates that 58.5% of all train accidents occur at a speed less than 30 miles an hour, 32% occur at a speed between 30-50 miles an hour, 9.4% occur between 50-70 miles an hour, and 0.1% occur at speeds exceeding 70 miles an hour.

Fires other than those involving ruptured tank cars of flammable liquids are unlikely to last longer than 1/2 hour, due to lack of sufficient fuel. Data relating major fires to train speed are sparse. It is estimated that 1.5% of all rail accidents involve fire of which 85% last less than 1/2 hour, 14% last between 1/2 hour and 1 hour, and 1% of the fires last more than 1 hour.

The probabilities for rail accidents are listed in Table 3.

Accident Statistics for Barges

Records for fiscal year 1970 for domestic waterborne traffic[45] show a total of 506 billion ton-miles of water traffic with 548 cargo barge accidents reported. Data are not available to indicate the fraction of those ton-miles due to barge traffic. We estimated the total barge ton-miles to be 380 billion. According to the Coast Guard report, miscellaneous types of vessels, including cargo barges, were involved in accidents which resulted in 33 injuries and 33 fatalities during that period.

The available data can not be analyzed in the same way as the data for rail or truck transport. On the basis of discussions with the U. S. Coast Guard, it is assumed that the average net (cargo) weight of a typical barge is about 1,200 tons. The total number of barge-miles would then be about 310 million. This yields an accident rate of about 1.8 accidents per million barge miles.

There are very few data available on the severity of accidents involving barges. Barges travel only a few miles per hour; therefore, the velocity of impacts in accidents would be small. Because of the large mass of the vehicle and cargo, severe impact forces could be encountered by packages (spent fuel casks) aboard barges. A forward barge could impact on a bridge pier and suffer crushing forces due to other barges being pushed into it. A coastal or river ship could knife into a barge. Fires could result in either case. An extreme accident, i.e., an extreme impact plus a long fire, is not considered credible. The likelihood of a severe fire in barge accidents is small because of the availability of water at all times. Also, since casks could be kept cool by sprays or submergence in water, loss of mechanical cooling can be compensated for.

The likelihood of cargo damage occurring in a barge accident is much less than in the case of rail accidents. For purposes of this analysis, and based on U. S. Coast Guard data, it is estimated that about 90% of the barge accidents would result in minor or no damage to the cargo, and would not involve fires. Moderate cargo damage due to impact would result in 8% of the barge accidents and severe damage in 2%. Fire would be likely only in those accidents involving moderate or severe cargo damage, and it is estimated that the likelihood of a fire in severe accidents would be 10 times that in moderate accidents. Based on the 1970 data, with only one cargo fire reported, it is estimated that fire would occur in 0.65% of the moderate accidents and 6.5% of the severe accidents.

There are no data on the duration of fires in barge accidents so we have used the rail figures of 85% of all fires lasting less than 1/2 hour, 14% lasting between 1/2 and 1 hour, and 1% lasting more than 1 hour.

The probabilities for barge accidents have been incorporated into Table 3.

Accident Severity Categories

In Table 2, accidents are categorized by degree of severity in terms of velocity of vehicle impact and incidence and duration of fire.

TABLE 2

Accident Severity Category	Vehicle Speed at Impact (mph)	Fire Duration (hr)
1. Minor	0-30	0-1/2
	30-50	0
2. Moderate	0-30	1/2-1
	30-70	<1/2
3. Severe	0-50	>1
	30-70	1/2-1
	>70	0-1/2
4. Extra Severe	50-70	>1
	>70	1/2-1
5. Extreme	>70	>1

Accident Probability

Table 3 shows the probabilities of an accident in each of the five accident severity categories and for each of the three modes of transport calculated on the basis of the data presented earlier.

From Table 3, we see that the differences between the truck, train, and barge accident probabilities in terms of accidents per mile in each of the severity categories are small. For purposes of estimating the risks

TABLE 3

ACCIDENT PROBABILITY

Severity Category	Vehicle Speed (mph)	Fire Duration (hr)	Probability per Vehicle Mile: Rail	Truck	Barge*
Minor	0-30	<1/2	$6x10^{-9}$	$6x10^{-9}$	--
	0-30	0	$4.7x10^{-7}$	$4x10^{-7}$	$1.6x10^{-6}$
	30-50	0	$2.6x10^{-7}$	$9x10^{-7}$	$1.4x10^{-7}$
Total			$7.3x10^{-7}$	$1.3x10^{-6}$	$1.7x10^{-6}$
Moderate	0-30	1/2-1	$9.3x10^{-10}$	$5x10^{-11}$	--
	30-50	<1/2	$3.3x10^{-9}$	$1x10^{-8}$	$8x10^{-9}$
	50-70	<1/2	$9.9x10^{-10}$	$5x10^{-9}$	$2x10^{-9}$
	50-70	0	$7.5x10^{-8}$	$3x10^{-7}$	$3.4x10^{-8}$
Total			$7.9x10^{-8}$	$3x10^{-7}$	$4.4x10^{-8}$
Severe	0-30	>1	$7.0x10^{-11}$	$5x10^{-12}$	--
	30-50	>1	$3.9x10^{-11}$	$1x10^{-11}$	$9.3x10^{-11}$
	30-50	1/2-1	$5.1x10^{-10}$	$1x10^{-10}$	$1.3x10^{-9}$
	50-70	1/2-1	$1.5x10^{-10}$	$6x10^{-12}$	$3.3x10^{-10}$
	>70	<1/2	$1x10^{-11}$	$1x10^{-10}$	--
	>70	0	$8x10^{-10}$	$8x10^{-9}$	--
Total			$1.5x10^{-9}$	$8x10^{-9}$	$1.6x10^{-9}$
Extra Severe	50-70	>1	$1.1x10^{-11}$	$6x10^{-13}$	$2.3x10^{-11}$
	>70	1/2-1	$1.6x10^{-12}$	$2x10^{-13}$	--
Total			$1.3x10^{-11}$	$8x10^{-13}$	$2.3x10^{-11}$
Extreme	>70	>1	$1.2x10^{-13}$	$2x10^{-14}$	--
Total			$1.2x10^{-13}$	$2x10^{-14}$	--

* Barge accident probabilities are based on the duration of the fire and actuarial data on cargo damage. The impact velocities of all barge accidents were considered to be less than 10 mph, but for the purposes of this table, minor cargo damage is assumed to be equivalent to vehicle impact speeds of 0-30, moderate cargo damage 30-50 and severe cargo damage 50-70.

in this analysis, a single value rounded off to one significant figure is taken for all three modes of transport as shown in Table 4.

TABLE 4

Accident Probabilities for Truck, Rail, and Barge per Vehicle Mile for the Accident Severity Categories

Minor	Moderate	Severe	Extra Severe	Extreme
2×10^{-6}	3×10^{-7}	8×10^{-9}	2×10^{-11}	1×10^{-13}

Unusual Accidents

Certain other accident circumstances can be postulated which may have a damaging effect on a package and for which the probability depends on other than the number of miles traveled.

1. Landslides. If an irradiated fuel cask is covered in a landslide such that it is unable to dissipate its heat, the temperature in the container will continue to rise until the container reaches equilibrium or is removed from the insulating surroundings. The probability that an irradiated fuel shipment would be present on a truck or railcar which is involved in a landslide and the irradiated fuel cask covered with dirt in a manner such that very little of the heat can be dissipated is believed to be extremely small.

2. Immersion in Water. Because very few accidents and few transshipments involving shipments of fuel or radwaste are expected to occur over water, it is extremely unlikely that a package of fuel or radwaste would be accidentally dropped into water. If dropped into shallow water, the package is unlikely to be damaged. In most cases, a package, cask or drum dropped into deep water would leak inward, through a gasket or valve, so the external and internal pressures would equalize as the package, cask, or drum sinks.[48,49,50] In some cases, the container might collapse. Some small amounts of radioactive material might be released. The container would seek the lowest level possible, either at the bottom or at a flotation level if the contents were low-density materials and remain at that level until recovered, or until dissolved by the corrosive effects of the water over many years.

The heat from a cask of irradiated fuel immersed in water would be released to the water. In most cases, suitable recovery procedures could be implemented in a reasonable length of time to remove this thermal heat source from the body of water. For this reason and because such an accident has such a low probability of occurrence, the heating of the water in such an accident is too small to justify quantitative evaluation.

3. Human Error. The adequacy of the design of a container can be compromised by an error on the part of the person loading and closing the package. One or more bolts may be left out or not properly tightened, a gasket misplaced or omitted, or a brace or "holddown" piece left off. The chances of such an error are small because of the procedures required by the regulations for examination of the closed container prior to each shipment, including tests for leak tightness, where necessary.

   Use of the wrong materials or errors made in construction also can result in a container failing to function properly during transport. The requirements imposed by the regulations on container manufacturers and shippers reduce the likelihood of such errors not being corrected prior to use.

   Each year a few packages are reported to have leaked even though not involved in an accident (e.g., the Delta Airlines incident of December 31, 1971), perhaps 8 out of 800,000. Many of these incidents are believed to be due to human error in closing the container. Perhaps 1 in 10 improperly closed packages is detected and reported. These usually involve shipments of liquids or gases and the amount of leakage is small. For such containers, Type A packages, it is estimated that 1 in 10,000 shipments is improperly closed when shipped.

   Taking into account the size of the components in most Type B packages, e.g., casks, and the attention to detail required in the closing procedures for casks and other Type B packages, it is estimated that 1 in 100,000 type B packages, including irradiated fuel casks, may be improperly closed when shipped.

Relationship of Accident Severity to Package Damage

The amount of damage to a package in an accident is not directly related to the accident severity; that is, in a series of accidents of the same severity, or in a single accident involving a number of packages, the amount of damage to the packages involved may vary from no damage to extensive damage

Various factors limit the effect accident conditions will have on a package.[51,52] In relatively minor accidents, serious damage to packages can occur due to impacting on sharp objects or by being struck by other cargo. Conversely, in extreme accidents, damage to some packages may be minimal. In some cases, the packages may be thrown free of the impacting vehicles or be so located in the vehicle that they are unaffected by the impact or the fire that ensues. Package damage depends on the form and amount of energy sustained by the package and the ability of the package to withstand those forces. The form and amount of the energy transmitted to the package in an accident depends on several factors which vary according to the accident circumstances.

The ability of a package to withstand accident forces depends on the design of the package and the quality assurance exercised in its manufacture, use, and maintenance.

DOT and AEC regulations specify certain package accident damage tests[53] which provide a means for reproducing in the laboratory or in the field the same general type and degree of damage a package might reasonably be expected to sustain in a severe transportation accident. Any package which can be shown to meet those standards is called a "Type B" package and can be expected to withstand accidents without leakage or significant shielding loss. The tests do not in themselves represent a transportation accident.

There are four such tests. They are a 30-foot freefall onto a flat unyielding surface, a 40-inch freefall onto a steel plunger, a thermal test and immersion in water. To better understand the design requirements imposed by the accident damage test criteria, the 30-foot freefall and the thermal test are discussed in some detail.

Although the velocity at the time of impact in the drop test is about 30 mph, the test requires dropping the package, including the protective shield if it is part of the package, on an unyielding surface. In very few accidents does the vehicle impact with an unyielding surface. In a real accident, the forces the package sustains are mitigated by the angle of impact of the vehicle, the crushing of the vehicle, which absorbs much of the impact, and the fact that, for impacts of heavy objects such as transporting trucks, the object with which the truck collides in most cases yields and thus absorbs some of the impact.

For example, in an instrumented full-scale test of a 15-ton cask on a semi-trailer in which the trailer was driven into an immovable barrier at 28.5 miles per hour,[35] the cask received only a fraction of the stress it was designed to withstand. The cask remained tied in place on the trailer and was undamaged, while the tractor was completely demolished.

As part of that same test series, a semi-trailer truck loaded with several different types of drums was driven into the immovable target at 42 mph. Several of the drums lost their lids but none of the inner containers was released or opened. About 50% of the drums were not damaged.

With respect to fire, the package must be designed to withstand the thermal test in which the package is subjected to the heat input from a radiant environment having a temperature of 1475°F and an emissivity of 0.9 for 30 minutes.

Severe transportation fires seldom last more than 1/2 hour, except in ships and storage depots,[54] because either the fuel is exhausted or the fire is estinguished by fire fighting crews. Although flame temperatures of liquids such as jet fuel or kerosene may reach 1800°F-2000°F, such peak temperatures are reached only very locally on the surface of material involved in the fire Only under very unusual circumstances is more than 50% of a package surface likely to be exposed to the flame for as long as 1/2 hour. Even in a longer fire, the package may be in a location where the fire has little or no effect on it.

For the above reasons, it is concluded that a package designed to meet the thermal test requirements in the regulations as a Type B package is likely to withstand the fire conditions in transportation accidents.

Type A Packages (e.g., drums of low level radwastes)

It is unlikely that a Type A package will be damaged and very unlikely that it will be breached in a minor accident. Based on experience and tests,[35] some fraction, perhaps 10%, of Type A packages will not be breached in very severe accidents.

Type B Packages

Based on regulatory standards and requirements for package design and quality assurance, results of tests, and past experience, Type B packages are likely to withstand all but very severe, highly unusual accidents. The probability of a Type B package being breached is low, so low that detailed consideration is not required in this analysis. Although the consequences of a release could be serious, the probability of occurrence is small, and therefore the risk or impact on the environment is very small.

APPENDIX B - SECTION I

CONSEQUENCES OF TRANSPORTATION ACCIDENTS INVOLVING SHIPMENTS OF NUCLEAR FUEL OR SOLID RADIOACTIVE WASTE

Estimates of Releases in Accidents

Estimates of the amount of radioactive material released in the unlikely event that a container is breached are given in this Appendix, taking into account engineering assessments of a variety of package designs, actual accident experience, the properties of the fuel and radwaste, and experience in shipment, reactor operation, and storage. In the case of Type B packages, the accidents analyzed which exceed the design basis accidents are practically incredible.

The mechanical and physical effects the accident forces would have on the contents, i.e., the fuel rods and solidified or compacted waste, and on the rate and amount of release when a breach of containment occurred, were considered in estimating the release in each type of accident. Consideration also was given the influence of the accident forces on dispersion of the released material. The consequences in terms of potential doses to people were calculated for the estimated releases of krypton-85, iodine-131, and fission products. Normal distributions of weather and population densities for a release on land were used in the calculations.

The overall probability of release into water is smaller than release on land because, with the exception of barge transportation, most of the transportation occurs over land.

The consequences of a release into water would depend on the characteristics of the material released and the conditions of use of the water. The release into water could affect soluble materials, and very little of the solid radwaste and none of the nuclear fuel is shipped in soluble form. With respect to release of fumes or dust, if the material is not soluble, the potential exposure levels would probably be smaller since dispersion in water would result in dilution. For dusts or fumes, even if soluble, the limits on the concentration in air are more restrictive than the limits on the concentrations in water. Also, if desired, depending on the circumstances, some restrictions on the use of contaminated water could be imposed.

Assumptions as realistic as the state of knowledge permits were used in estimating the consequences of accidents. Wherever possible, realistic average values were used; otherwise pessimistic assumptions were made. For example, in estimating exposures in an accident, ground level releases were assumed. The rise of the heated plume in a fire was not considered, although in most cases that would reduce the effects. The distribution of population density in the Eastern United States as projected by the Staff for 1980 was taken as representative of the population densities along routes on which the shipments will travel.

For analysis of accidents, random distribution of the population was assumed; that is, it was assumed that an accident may occur in each population density with a probability equal to that with which that density is found in the distribution. In general, however, the probability of an extremely severe accident is less in the higher populated areas owing to generally lower vehicle speeds and, for rail transport, better maintained roadways.

Some accidents in transportation may produce stresses on packages more severe than the stresses the packages are designed to withstand. The consequences of such accidents could be serious but the probability of occurrence of such accidents is extremely low. Quality assurance for design, manufacture, and use of the packages; continued surveillance and testing of packages and transport conditions; conservative design of packages; and the low probability of occurrence make the environmental risk from such accidents extremely low.

For this analysis, the present methods of packaging, ways and means of transportation, traffic patterns, etc., have been assumed to continue in use for the projected period of operation of the reactor.

The values of package damage chosen are related to the present level of design requirements in the packaging standards and criteria. Should the standards be lowered, the fractions of packages damaged in all types of accidents would be expected to increase, although the exact relationship would be difficult to predict. If the standards were increased, the fraction damaged would be expected to shift downward. Since the damage to the package does not depend directly on the severity of the accident, adding structural strength or stress resistance to the design would not be expected to reduce damage in direct proportion to the added strength. Furthermore, the added strength may increase the risk from common causes due to increased weight and number of shipments.

Based on consideration of the quantity and form of radioactive material in the package, postulated accident conditions, and certain other factors, the following estimates were made of the number of curies, Q, of radioactive material which might be released from a damaged package.

For any set of accident conditions which causes a breach in the container, a range of amounts of radioactive material might be released; that is, the quantity might range from nothing to a significant fraction of the contents. The amount estimated to be released, Q, represents the most probable maximum release for that set of accident conditions. An amount greater than Q is considered to be less likely to be released under the same conditions than the estimated amount.

If sufficient data were available, the probability of release, P_R, for each release could be calculated based on the probability of an accident of a particular severity occurring, the probability of a package being breached in such an accident, and the probability of the release occurring with the package breached.

Using the calculative method outlined in Section II of Appendix B, the probability of release, P_R, and amounts of radioactive release, Q_O, could be used to derive estimates of the probability that N or more persons would receive a radiation dose of D or more mrem in a transportation accident.

The calculative procedure requires a determination of the probability of one or more persons receiving a specified dose for each of the accidents postulated. A summation of the probabilities for all of the accidents in a spectrum of accidents would provide an estimate of the overall probability of one or more persons getting a dose of D or more mrem from all accidents.

The spectrum of accidents should include the entire range of credible accidents up to the point that either the probability or the consequences of other accidents is so small that they would be unlikely to affect the value calculated for the postulated range of accidents.

Unirradiated Fuel

Because of the low level of radioactivity in unirradiated fuel, the design of packaging for unirradiated fuel is not required to be as rugged as the design of packagings for higher levels of radioactivity, and therefore is more susceptible to damage in an accident. The form of the unirradiated fuel, i.e., high-density, high-melting point pelletized uranium oxide contained in sealed zircaloy tubes, makes the dispersion of any of the oxide extremely unlikely even in the event of severe damage to a package of fuel. The radioactivity of the oxide is very low. Even if some dispersion were to occur, the radiation doses would be very small. Except for an accident resulting in nuclear criticality, the radiological impact on the environment from accidents involving unirradiated fuel is negligible.

Containers for shipping unirradiated fuel are required to be designed to prevent accidental criticality under normal and accident conditions. Considering the practical conditions required for achieving criticality (viz., release of a number of fuel elements from their respective containers, assembly of these elements in a close array and moderated, e.g., with water in and around the assembly), the probability of criticality being achieved in an accident is extremely small. If such an accident should occur, the consequences would be mitigated by having taken place in a moderator such as water which acts as both a radiation shield and an absorber of some of the gaseous fission products which might be released.

The consequences of postulated accidents involving unirradiated fuel shipments are summarized below:

1. Normal conditions--nothing released even if the lid is loose.

2. Accidents--nothing significant released except in unusual circumstances; e.g.,

   a. Fuel element is knocked out of a package and run over by a train. It is unlikely that contamination of other than localized areas would occur; no significant airborne contamination would be expected.

   b. Accidental criticality. Consequences:

      In the unlikely event of accidental criticality, the critical array likely would be quickly disassembled by pressures developed during the reaction but a nuclear explosion is impossible. The critical reaction would last only a few seconds and probably would not recur. It is estimated from 10^{17} to 10^{18} fissions might take place[16] but this would not be expected to cause release of any radioactive materials from the fuel elements. Residual radiation levels due to induced radioactivity in the fuel elements might reach a few rem per hour at 3 feet.

      Persons within a few feet of such a critical assembly would receive a lethal dose of gamma and neutron radiation unless shielded by intervening material. Persons beyond 100 feet would be unlikely to receive serious radiation exposures; the cumulative dose to the 7500 persons located within 1/2 mile of the incident but beyond 100 feet is estimated to be no more than 500 man-rem. The consequences would be reduced because the reaction takes place in a moderator such as water which acts both as a radiation shield and an absorber of some of the gaseous fission products if any were released. Recovery of the fuel elements and cleanup of the immediate area would be required.

Irradiated Fuel

Irradiated fuel is packaged in large, rugged containers, frequently with liquid coolant, because of the high radiation levels and heat output. At the time of shipment, the irradiated fuel will have been "cooled" about 150 days, on the average. The total radioactivity in the fuel will be approximately 4×10^6 curies per metric ton of irradiated fuel.

Measurements made in reactor operation show that no activity other than some gases will be released from intact fuel assemblies. That means that until the fuel cladding is broken or perforated, only the surface contamination on the fuel assemblies (activation and corrosion products, mostly Co-60 and Fe-59) would be expected to be present outside of the fuel cladding.

If the cladding of a fuel rod is penetrated, some of the radioactivity from inside the fuel rods may be released. The staff estimates all of the free gases in the void spaces and a fraction of the semi volatile and a smaller fraction of the non-volatile fission products and actinides might be released. Table 2 gives estimates of the activities in gaseous or other mobile form in the fuel rod void spaces which would be available for release from the fuel rods if the cladding were broken or perforated. The gases of significance are Kr-85, Xe-131m, and I-131.

Because of the regulatory limits in 10 CFR 71.35 on the radioactivity in the cask coolant, any fuel assembly which is releasing a significant amount of radioactivity must be placed in a separate, sealed container (i.e., "canned") prior to being loaded into the cask for shipment. Fuel assemblies releasing significant amounts of radioactivity while in the reactor will have been identified before being discharged from the reactor but some so-called "failed fuel" may g undetected. In the case of "failed fuel," much of the radioactivity in the fuel rod void space may have been released during the time the assembly remained in the reactor after failure and while stored in the canal for cooling prior to shipment.

It is believed conservative to assume that, under normal conditions of transport, 0.25% of the free gases and other activities from the fuel rod spaces would be outside the fuel assemblies in the cask coolant or cask cavity, in addition to the surface contamination mentioned above. Some residual contamination from the storage pool might also remain in the cask cavity and hence the coolant, since the loading operation is carried out in the storage pool water.

Under normal conditions the primary coolant, that is, the coolant which is in contact with the irradiated fuel in the cask, may be contaminated but the level of contamination will be small. Based on recent experience reported at the Savannah River Plant, the activity in water-filled casks ranges from 10^{-4} to 10^{-2} μCi/cc. For the higher burn-up power fuel, a level of 1 μCi/cc has been estimated. The activity may include a mixture of activation, corrosion, and fission products.

The total amount of activity in the coolant based on 10^6 cc of coolant in a rail cask and 10^5 cc in a truck cask would be 1 Ci and 0.1 Ci, respectively. For purposes of estimating releases in accidents, that activity is assumed to be present in the coolant in the form of fission products. Under normal conditions, that activity would be present in addition to the Kr-85 and I-131 released from the air gap in the fuel due to perforations in the cladding of a small fraction of the rods. From preliminary analyses, it appears that it would require a severe impact, probably in excess of 50 miles per hour, to cause fuel rods to rupture. When accident conditions result in perforation of a greater percentage of the rods, additional fission products are assumed to be released as indicated in Table 2.

Most casks have a pressure relief system which is expected to vent when the internal pressure exceeds a preset level. At present, the systems are usually designed to reseal after the excess pressure is relieved.

TABLE 1

Basic Estimates - Irradiated Fuel

0.5 MT irradiated fuel per cask for truck

3.2 MT irradiated fuel per cask for rail

1 cask per truck or rail car

60 truck shipments per 1100 MWe reactor-year

10 rail shipments per 1100 MWe reactor-year

[illegible]00 miles shipping distance from power plant to fuel recovery plant.

Percentage of material released from irradiated fuel cask which becomes airborne:

100% of gases (krypton & iodine)

1% of the contaminants in the coolant in the absence of fire and 10% if fire is present.

TABLE 2

FUEL ROD VOID SPACE ACTIVITY

Type of Radio-active Material	Total Inventory[16] 150 days cooling curies/metric ton	% in void spaces of fuel rods*	Activity in void spaces, curies/metric ton
Kr-85	1.12×10^{4}	30	3.4×10^{3}
I-131	2.17	2	4.3×10^{-2}
Other fission products	4.39×10^{6}	0.01**	400
Actinides, (Pu,Am,Cm,etc.)	1.36×10^{5}	essentially none	neg.***
Xe-131m	3.27	2	0.1***
I-129	2×10^{-3} [20]	30	6×10^{-4}***
H-3	6.92×10^{2}	1	7***

* Realistic gap activities in terms of percent of total inventory prepared by AEC's Directorate of Licensing based on references 20 through 32.

** A conservative (high) value estimated on the basis of leaching the outer 1.2×10^{-5} inches from the surface of the uranium oxide fuel.

*** Due to the small amounts present, the dose contribution from Xe, I-129, H-3, and the actinides may be neglected compared to the doses from the other radionuclides.

TABLE 3

ESTIMATED RELEASES FROM RAIL CASKS
UNDER UNUSUAL ACCIDENT CONDITIONS

		Q, Activity Released* (in curies)		
		Kr	I-131	Fission Products
I.	Undetected Leak:	-	-	4.5×10^{-4}
	coolant released at a rate of 0.001 cc/sec; 450 cc in 5 days			
II.	Overpressure	0.03	3×10^{-7}	1×10^{-3}
	Pressure relief valve operated 0.1% of coolant released 0.25% of fuel rods perforated			
III.	Overheated	30	3×10^{-4}	1
	All coolant released			
IV.	Assume 30% of fuel rods perforated -- all coolant released	5.5×10^{3}	0.1	650

* Based on the rail cask containing 3.5 metric tons of fuel. Equivalent releases from truck casks carrying 0.5 metric tons of fuel would be about 1/7th the activities shown except for the undetected leak which would be the same as shown.

Appendix "B"

In one design of rail cask now under evaluation (GE, IF-300),[55] complete failure of the external cooling system will cause the cask to overheat over a period of several hours. In that case, under certain adverse but unlikely conditions, the temperature of 50% of the fuel elements would reach 1200°F, which could cause perforation of the cladding on some of the rods if the elements were of the present PWR type. According to the analysis, the present BWR type of elements would not be expected to perforate.

Truck casks are not expected to reach rod perforation temperatures except under an extended fire condition.

Four examples of postulated accidents involving irradiated fuel casks are given below.

Example 1. A rail cask containing 3.2 MT of irradiated fuel is in an accident involving a severe impact and fire which causes a breach in the containment. If 10% of the rods were perforated and 100% of the coolant released, as much as 1.1×10^3 Ci of Kr-85, 1×10^{-2} Ci of I-131 and 130 Ci of gross fission products could be released.

The consequences of this type of accident were estimated assuming a ground-level release under average weather conditions with all of the krypton and iodine and 1% of the gross fission products being dispersed in the air. Because of the severity of the accident and the precautions taken immediately afterward, persons are not expected to be closer than 50 meters downwind from the accident, the direction in which the highest exposures would occur.

A cumulative whole-body dose of about 0.4 man-rem from the Kr-85 would be received by the million people nearest the accident, assuming 10^4 persons per square mile. Persons 50 meters downwind could receive doses as high as those given in the Table 4.

The contamination on the ground, assuming the coolant is released as vapor and the contamination dispersed, would result in Range I levels, requiring decontamination according to standards[56] of the Environmental Protection Agency, over an area of about 3000 square feet and Range III levels, requiring further consideration as to whether specific action would be required, over an area of about 0.1 square mile. For a high population density of 10,000 persons per square mile, only one person must be evacuated in the 3,000 square foot area that is contaminated; the cost of evacuation and contamination cleanup is estimated to be $10,000 to $50,000.

TABLE /

CALCULATED DOSES FROM RAIL ACCIDENT

	Organ	Centerline Dose* (rem)	Average Dose* (rem)
Kr-85	Skin	1.2	0.06
	Bone marrow, gonads, lens of the eye	0.02	8×10^{-4}
I-131	Thyroid	0.02	1×10^{-3}
Gross Fission Products	Bone	6	0.3
"	Lung	8	0.4

* The radioactive material would be distributed downwind from the accident so that the isopleth (i.e., boundary lines of equal doses) would be cigar-shaped. The centerline dose is the dose which might be received by a person on the centerline of that pattern at a distance of 50 meters from the accident and the average dose is the average of the doses to all persons at 50 meters in all directions from the accident.

The consequences of the accident described in this example also were estimated using the method outlined in Section II of this Appendix. The probabilities of N or more persons receiving doses of D or more millirem as a result of a release of 1.1×10^3 Ci of Kr-85, 1×10^{-2} Ci of I-131, and 130 Ci of gross fission products, with all of the krypton and iodine and 1% of the gross fission products being dispersed in the air, were calculated. The values for P_N/P_R are given in Table 5 through 8.

The number of rail shipments of irradiated fuel from a reactor is estimated to be 10 per year. For a shipping distance of 1,000 miles, that makes a total of 10,000 shipping miles per year. The probability of a shipment being involved in an extra-severe accident in transport is 1×10^{-11} per vehicle mile (see Appendix A). Based on the accident data available, the standards for design of the package and results of package tests, we estimate no more than 1 in 10 packages involved in an extra-severe accident would be damaged to the extent that a release of the magnitude specified could occur. Based on these numbers, the probability of a release (P_R) of the magnitude specified would be approximately 1×10^{-8} per reactor year from a transportation accident involving irradiated fuel.

If the probability of the release occuring is taken to be 10^{-8} per reactor year, the probabilities (P_N) of N or more persons receiving doses of D or more millirem per reactor year from the rail transportation of irradiated fuel would be the probabilities in Tables 5 through 8 multiplied by 10^{-8}. That is, each value given in the tables for (P_N/P_R) should be multiplied by (P_R) to obtain the probability (P_N).

As shown in the Tables 5 through 8, even if the probability of a release were substantially higher than 10^{-8}, the probability of a significant exposure as a result of releases of the magnitude assumed would still be small.

Example 2. Some designs of rail casks have an external mechanical cooling system. An accident may cause moderate damage to the cask such that the mechanical cooling system becomes inoperative. If no corrective action is taken and the ambient temperature is above 100°F, the temperature of the fuel in the cask will increase enough in a few hours to cause an overpressure in the cask cavity, and some of the coolant will be released through the vent system. This also may occur in some cask designs if the cask is involved in a severe fire.[57]

Venting may occur in a series of releases; one design permits about 5% of the gas in the cask cavity to be released at a time. The activity released would be quite small, amounting to perhaps 5% of the total activity in the coolant. That design contains approximately 2.3×10^6 cc of water. For a contamination level of 1 microcurie per cc, the total activity released would be about 0.1 curie of primarily cesium-137.

<u>Example 3</u>. The rail cask in Example 2 is left unattended for several hours. The temperature of the fuel in the cask will continue to increase until adequate means are provided for dissipation of the heat. In a matter of several hours, some of the fuel may reach a temperature at which the cladding will perforate. Perforation is due to overpressure of gases in the air gaps and weakening of the cladding due to increased temperature. For example, in one rail cask design if the mechanical cooling system is inoperable and the ambient temperature remains at or near 130°F for at least 11 hours, the designer estimates 50% of the fuel rods may reach 1200°F, which is the perforation temperature for PWR fuel rods. Under the same conditions, BWR fuel elements would not be expected to reach perforation temperature.

The likelihood of a cask remaining unattended after loss of mechanical cooling or after being involved in a serious fire for a period long enough that overheating would be expected can be reduced by appropriate administrative controls such as escorts, alarming the mechanical cooling system, inspection of the shipment at regular intervals, and notification of the shipper in case of any failure of mechanical cooling or involvement in an accident. Where considered important, shippers may be required to establish and implement such procedures.

The radioactivity released in such an accident could be as much as 5.5×10^3 Ci of Kr-85, 0.1 Ci of I-131, and 30 Ci of gross fission products.

<u>Example 4</u>. Perhaps an accident results in the cask being covered with dirt and debris in a landslide or dumped into a pile of soft dirt or other cargo so that the cask would be unable to dissipate all of the heat generated by the fuel. Under most circumstances, the cask would be removed before reaching excessive temperatures, and the accident would produce no adverse consequences other than cost of recovery. However, the temperature of the container would continue to rise until the container reached equilibrium or was removed from the insulating surroundings. If a rail cask were not removed, the releases could equal those postulated above for the loss of mechanical cooling.

<u>Release of Irradiated Fuel Elements</u>

Considering current cask design practices, it is improbable, but not impossible, that a cask could be damaged to the extent that one or more fuel elements would be released from the cask. The methods of installing and securing cask closure devices are such that the closure device is not likely to be opened or removed in any accident. Release of a fuel element is unlikely except in an extremely severe accident in which unusual circumstances cause rupture of the cask.

If seven irradiated fuel elements were released from a cask in an unusual accident, the radiation level at 100 feet could be as much as 10^4 r/hr. Assuming the fuel elements remained unshielded for 10 hours, approximately 30,000 persons within a mile radius (based on 10^4 persons/square mile) might receive a cumulative dose of about 1000 man-rem. If a person remained unshielded at an average distance of 100 feet from the fuel elements for 6 minutes, he might receive a dose of as much as 1000 rem. Persons remaining near the exposed fuel for any appreciable length of time may receive large doses of radiation. Someone at a distance of 10 feet from the exposed fuel for about a minute, would receive a dose of 1000 rem. Remote equipment would be required to erect a shield around the fuel elements or to place them in a shielded box or to repackage them.

Relationship of Releases to 10 CFR Part 71 Limits

The amounts of radioactivity estimated to be released from an irradiated fuel cask in the accidents postulated for this analysis differ from the amounts specified in the package design criteria in 10 CFR §71.36. The design criteria were derived on the basis of both safety and feasibility for a range of contents and container designs which had been identified at the time that rule was being developed.

The amounts of radioactivity estimated to be released in the accidents postulated in this analysis take into account the physical and chemical characteristics of the particular type of fuel under analysis (high burnup uranium oxide pellets) and were derived using measured and calculated values from operating experience in light-water reactors as well as in shipping of irradiated fuel.

Solid Radioactive Wastes

Estimates of probabilities and amounts of releases of solid radioactive wastes in accidents in transportation involve considerations different from those for irradiated fuel. The packaging for solid wastes includes both Type A and Type B packaging so that some of the packaging for waste is not expected to withstand the accident conditions.

The containment is provided by the form of the material (i.e., radioactive material bound on clothing, dispersed in concrete, or otherwise confined to some degree) and by the package--drums in most cases. The drums are expected to lose lids under accident conditions with probability equal to that estimated for a small breach of containment.

The form of the material ranges from compacted combustible materials to material which has been dewatered and solidified, in most cases as concrete. The radioactive contamination in compacted waste usually will not be in an available form if released in an impact; that is, pieces of contaminated clothing, etc., may be spread around, but the contamination is bound on the inert materials, such as clothing, and is unlikely to be released from the clothing unless burned or washed out by water. On the other hand, the contaminated concrete is not likely to be affected by fire, but some of the concrete may be shattered by a strong impact force.

The probability and extent of release from a package of solid waste is about the same whether the waste is transported by truck or by rail. The same types of packages are shipped by truck and by rail. The only difference is that more packages are carried on a rail car than on a truck. The probability of an accident of any of the defined degrees of severity is shown to be about the same for rail or truck per vehicle mile.

The number of miles traveled by truck is greater than that by train in proportion to the number of drums carried by each. Therefore, the probability of a load of drums being involved in an accident is greater by truck than by rail but the larger number of drums in the rail car balances the difference in terms of probability of leakage of a drum of waste.

TABLE 5

PROBABILITY OF N OR MORE PERSONS RECEIVING A DOSE TO THE SKIN OF D MILLIREM OR MORE FROM THE RELEASE OF 1100 CURIES OF KRYPTON-85 IN AN ACCIDENT

Number of People N	Dose (millirem) D 1	10	100	1000	5000
1	0.9	0.5	0.1	2×10^{-2}	3×10^{-3}
10	0.6	0.2	3×10^{-2}	1×10^{-3}	
10^2	0.2	4×10^{-2}	2×10^{-3}		
10^3	7×10^{-2}	2×10^{-3}			
10^4	1×10^{-2}				
10^5	5×10^{-4}				

* * * * *

TABLE 6

PROBABILITY OF N OR MORE PERSONS RECEIVING A DOSE TO THE THYROID OF D MILLIREM OR MORE FROM THE RELEASE OF 0.01 CURIES OF IODINE-131 IN AN ACCIDENT

Number of People	1	10	100	1000
1	0.5	9×10^{-2}	1×10^{-2}	2×10^{-4}
10	0.1	1×10^{-2}	4×10^{-4}	
10^2	2×10^{-2}	6×10^{-4}		
10^3	1×10^{-3}			

Appendix "E"

TABLE 7

PROBABILITY OF N OR MORE PERSONS RECEIVING
A DOSE TO THE WHOLE BODY OF D MILLIREM OR MORE
OVER A PERIOD OF ONE YEAR FOLLOWING THE RELEASE
IN AN ACCIDENT OF 130 CURIES OF GROSS FISSION PRODUCTS
WHICH DEPOSIT ON THE GROUND. 80% OF THE DOSE IS TO THE SKIN

umber of People N	Dose (millirem) D 1	10	100	1000	5000	10000
1	1	1	1	0.9	0.7	0.7
10	1	1	0.9	0.7	0.3	0.2
10^2	1	0.9	0.6	0.3	0.1	6×10^{-2}
10^3	1	0.7	0.4	9×10^{-2}	2×10^{-2}	6×10^{-3}
10^4	0.8	0.5	0.2	3×10^{-2}	9×10^{-4}	2×10^{-4}
10^5	0.7	0.4	8×10^{-2}	2×10^{-3}		

TABLE 8

PROBABILITY OF N OR MORE PERSONS RECEIVING
A DOSE TO THE LUNGS OF D MILLIREM OR MORE FROM
1.3 CURIES OF GROSS FISSION PRODUCTS RELEASED IN AN
ACCIDENT WHICH BECAME AIRBORNE

Number of People N	Dose (millirem) D 1	10	100	1000	5000	10000
1	1	0.8	0.3	5×10^{-2}	1×10^{-2}	4×10^{-3}
10	0.8	0.3	6×10^{-2}	4×10^{-3}	3×10^{-4}	4×10^{-5}
10^2	0.4	9×10^{-2}	6×10^{-3}	1×10^{-4}		
10^3	0.1	1×10^{-2}	2×10^{-4}			
10^4	4×10^{-2}	5×10^{-4}				
10^5	4×10^{-3}					

Appendix "B"

TABLE 9

Basic Estimates - Solid Radioactive Wastes

a. Soit solid wastes compacted in 55-gallon drums.

100 drums produced per 1100 MWe reactor year

1 curie of radioactivity per drum

2 truck shipments per year, 50 drums per truckload

1 rail car shipment per year; 100 drums per carload

1000 miles shipping distance from power plant to waste disposal site. If the waste burned in an open fire, it is unlikely that much of the activity would be widely dispersed. Most of the activity, perhaps as much as 99%, would remain in the ashes.

b. Resins, sludges, etc. dewatered and consolidated in 55-gallon drums.

3000 drums produced per 1100 MWe reactor year.

98% - Type A packages, limited to 20 curies/drum. About 3% low level compacted wastes and 95% average less than 0.3 curie per drum.

2% - Type B packages, 100 Ci maximum estimated activity per package; average estimated about 20 curies per drum.

60 truck shipments per year, 50 drums per truckload

20 rail car shipments per year; 150 drums per carload

500 miles shipping distance to waste disposal site.

Because of the form of the material, it is very unlikely that any significant amount of the activity in material burned in an open fire would be released, probably less than 10^{-5} of the activity in the contents.

Table 10 gives the estimated quantities of radioactive material which could be released in postulated accidents. The estimates are considered to be maximum values for the accident listed. Larger releases would be expected to have lower probabilities of occurring. The activity is expressed in curies of airborne fission products, although some other radioactive materials of lower degrees of toxicity would be present.

TABLE 10

ESTIMATED RELEASES FROM PACKAGES OF RADWASTE

Q - Activity in curies that become airborne

	Lid loose - one drum	Contents of 1 drum spilled out	Contents of 25 drums burned	25 drums broken open- severe impact
Compacted Waste	10^{-8}	10^{-6}	2.5×10^{-2}	2.5×10^{-5}
Type A Package	10^{-7}	10^{-6}	-	10^{-7}
Type B Package	10^{-6}	10^{-6}	-	0.25

Appendix "B"

APPENDIX B - SECTION II

POPULATION DOSE DISTRIBUTION PROBABILITIES

If radioactive material is released to the atmosphere in a short period of time at ground level and if it is assumed that there is no appreciable depletion of the airborne material, the dose caused by exposure to this material is

$$D = Q_0 K (X/Q) \qquad \text{(Rem)}$$

where Q_0 = curies released

K = dose coefficient $\frac{(\text{rem-m}^3)}{(\text{Ci-sec})}$ (These will be identified later)

Values of (X/Q) as a function of distance for ground level releases are given in Figure 1.

Values of isopleth areas A_{iW} (the area within which a particular dose D_i is equaled or exceeded) in square miles for selected values of the dose parameter $\frac{D_i}{Q_0 K} = \frac{(X)}{(Q)_i}$ are shown for Pasquill type weather conditions in Table 1 along with the weather probabilities and average wind speeds.

The number of people who receive a dose greater than D_i is proportional to the population density in the area involved. The probability of giving doses greater than D_i to N or more people is proportional to the probability of the release for a given weather condition being in an area with a population density m such that $m = N/A_{iW}$.

The fractional areas (F_m) with various population density ranges based on the populations within 50 miles of presently operating reactors calculated for the 1980 time period are given in Table 2. The table also gives the progressive summing of these fractions in two directions. This population distribution represents a relatively high average population density probably typical of the eastern United States. A distribution typical of the whole United States would be similar in shape but the fractional part with populations of 10,000 people or more per square mile would be about a factor of 10 less or about 0.001.

Given that a randomly located release has occurred, the probability of the release being in an area with less than m people per square mile is $\Sigma F(m)$. Conversely, the probability of the release being in an area with more than m people per square mile is $1 - \Sigma F(m)$. The population is assumed to be uniformly distributed around the scene of an accident, with a density of m persons/mi^2. The probability of any particular value of m is the same as for a random point in Eastern United States as projected by the Staff for 1980, based on data of the U. S. Bureau of Census and the results of a study of the 1980 projection of the population density distributions within 50 miles of 22 operating reactors. The function of m used for subsequent calculations is $P(>m)$, the probability that the population density exceeds m persons/mi^2. $P(>m)$ is given in Table 2 and in Figure 2. If the probability of release in an area with more than m people per square mile is defined as Pm, then $Pm = P_R (1 - \Sigma F(m))$ where P_R is the total probability of release in the selected zone. The value of Pm/P_R vs. m is shown in Figure 2. The partial probability of giving more than N people (where $N = mA_{1W}$) doses greater than D_1 for emission during a particular weather condition is given by

$$P_{NW} = P_W P_R \left[1 - \Sigma F(m)\right] \qquad \text{where } m = \frac{N}{A_{1W}}$$

The total probability is then

$$P_N = P_R \sum_W P_W \left[1 - \Sigma F(m)\right] \qquad \text{where } m = \frac{N}{A_{1w}}$$

The process of calculating the value of P_N/P_R is illustrated in Figure 3 for the case where $D_1/Q_0 K \geq 10^{-4}$. The individual partial probabilities for each weather condition are shown along with the total. Total values of P_N/P_R for other values of the dose parameter are given in Figure 4.

Values for the dose coefficient K are given in Table 3.

Figure 5 is a plot of values of D/KQ versus P_N/P_R taken from Figure 4. Given Q (the curies released in an accident), P_R (the probability of a release of that number of curies or more), and K (the dose coefficient), the probability of N or more persons receiving a dose of D or more mrem from that release can be determined.

The probability of N or more persons receiving a dose of D or more millirem per reactor year from transportation accidents is the sum of the probabilities of N or more persons receiving that dose from each accident in the spectrum of credible accidents.

TABLE 1 PASQUILL WEATHER TYPE DOSE ISOPLETH AREAS (A_{iW}) FOR SELECTED VALUES OF THE DOSE PARAMETER D/Q_0K, ASSOCIATED WEATHER PROBABILITIES (P_W) AND AVERAGE WIND SPEEDS ($\bar{U}_W$)

DOSE PARAMETER D/Q_0K	AREAS IN SQUARE MILES PASQUILL WEATHER TYPE						
	A	B	C	D	E	F	G
10^{-1}	5.8×10^{-6}	5.8×10^{-6}	4.6×10^{-6}	3.8×10^{-6}	1.6×10^{-5}	3.8×10^{-5}	1.9×10^{-4}
10^{-2}	6.2×10^{-5}	6.2×10^{-5}	5.0×10^{-5}	4.2×10^{-5}	1.8×10^{-4}	5.8×10^{-4}	2.1×10^{-3}
10^{-3}	5.8×10^{-4}	5.4×10^{-4}	4.2×10^{-4}	4.2×10^{-4}	1.9×10^{-3}	5.4×10^{-3}	3.1×10^{-2}
10^{-4}	5.0×10^{-3}	5.4×10^{-3}	4.2×10^{-3}	5.8×10^{-3}	2.3×10^{-2}	7.3×10^{-2}	4.6×10^{-1}
10^{-5}	3.5×10^{-2}	4.6×10^{-2}	4.6×10^{-2}	7.7×10^{-2}	3.3×10^{-1}	1.5×10^{0}	1.9×10^{1}
10^{-6}	1.5×10^{-1}	3.1×10^{-1}	6.2×10^{-1}	1.2×10^{0}	7.7×10^{0}	7.7×10^{1}	3.6×10^{3}
P_W	.019	.081	.136	.44	.121	.122	.081
$\bar{U}_W$ $\left(\frac{m}{sec}\right)$	2	3	5	7	3	2	1

TABLE 2 ESTIMATED POPULATION DISTRIBUTION
FOR THE 1980 TO 2000 TIME PERIOD

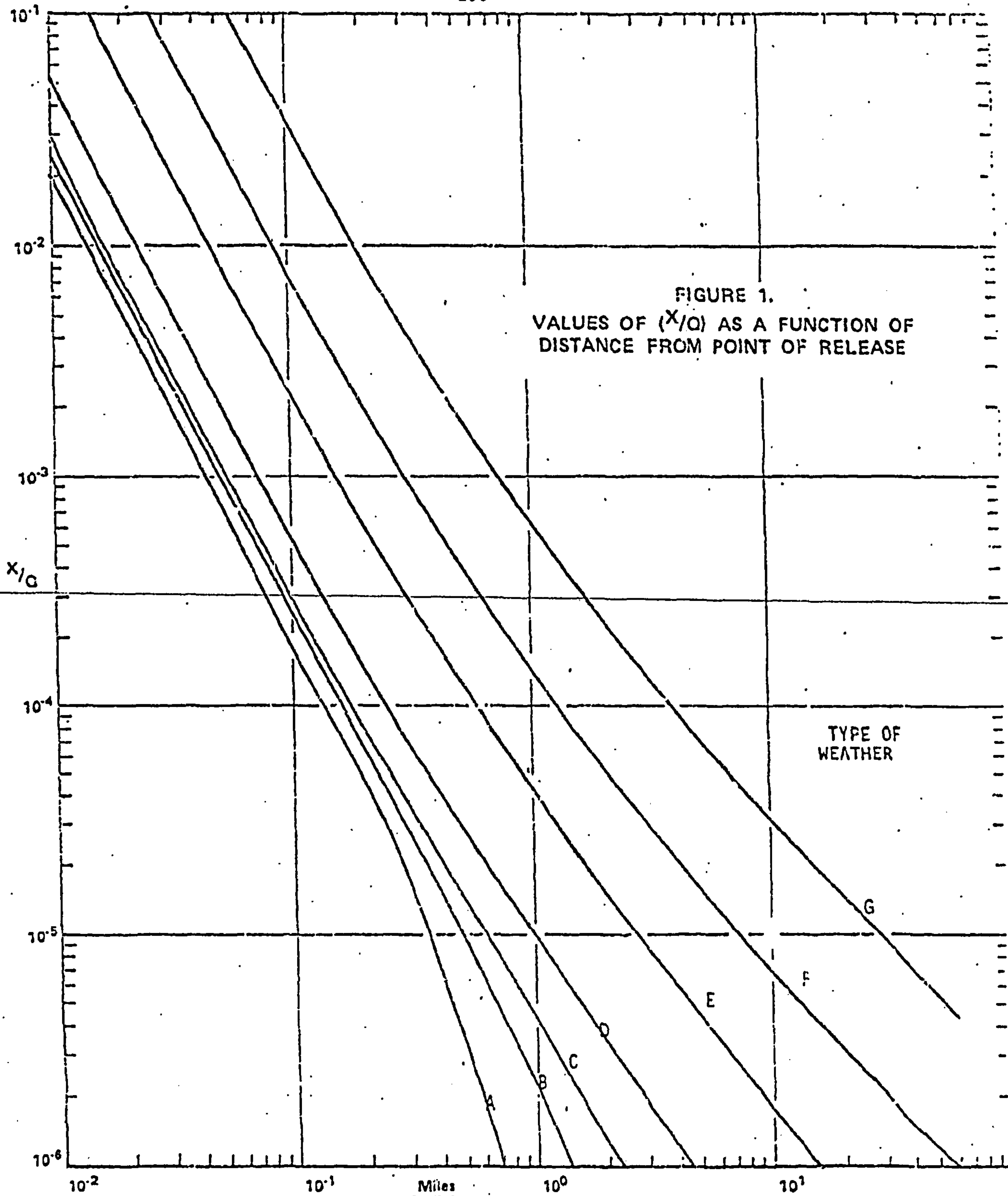
ZONE LIMITS - m (People/mile^2)	F_m	$\Sigma\ F_m$	$1 - \Sigma\ F_m$
0		0	1.000
	.255		
100		.255	.745
	.561		
1,000		.816	.184
	.174		
10,000		.990	.010
	.010		
>10,000		1.000	0

The estimated doses from iodine-131 releases are based on uptake from inhalation of contaminated air. The potential exposure from deposition of iodine on grass and uptake through the milk chain would be significantly below the levels of direct exposure for the accidental releases considered.

TABLE 3 VALUES OF

DOSE COEFFICIENT K

Radionuclide	Dose	K $\frac{(rem-m^3)}{(Ci-sec)}$
I-131	Thyroid - child dose due to inhalation	4.76×10^2
	- adult dose due to inhalation	3.18×10^2
Kr-85	skin - due to submersion in the cloud	0.053
Gross fission products (33,000 MWD/MT burnup, 30 MW/MT power level, 150 days cooling)	Whole body-(80% of which is skin dose) due to material deposited on the ground assuming no depletion of cloud. Exposure during first year after release, assuming no loss from ground.	7.30×10^2
	Lung-due to inhalation	1.11×10^2



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Appendix "B"

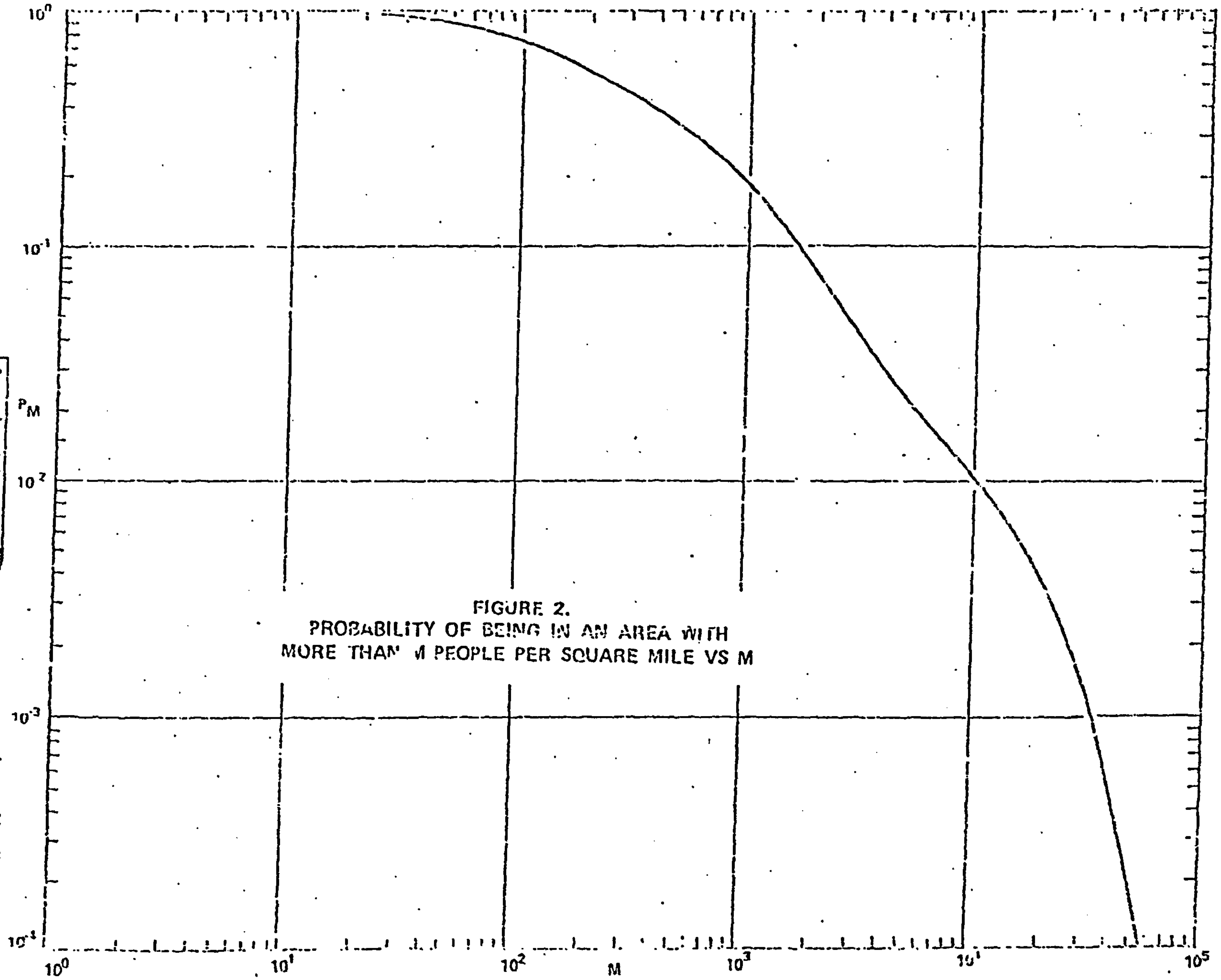

FIGURE 2.
PROBABILITY OF BEING IN AN AREA WITH
MORE THAN M PEOPLE PER SQUARE MILE VS M

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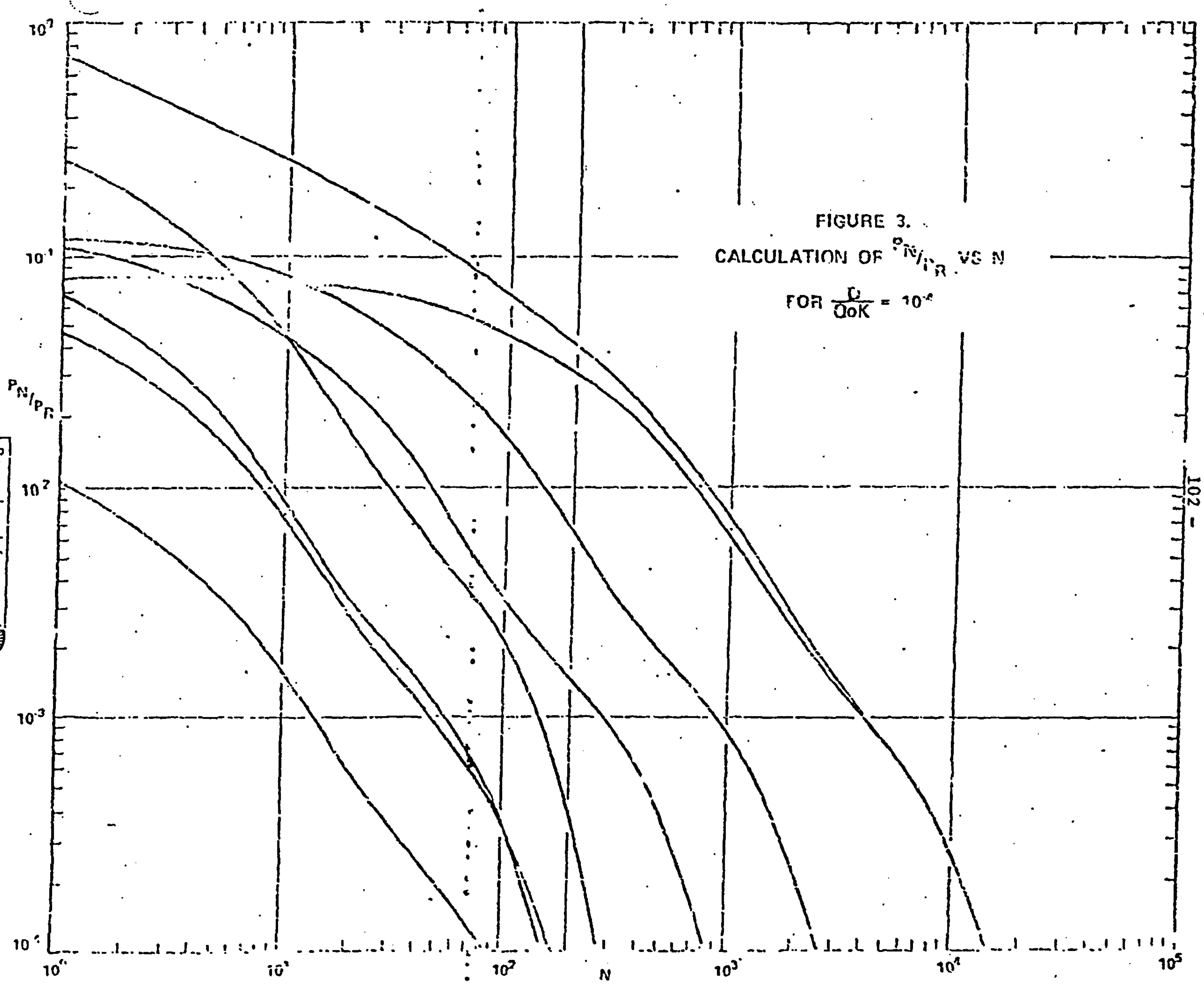
Appendix "H"



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Appendix "B"

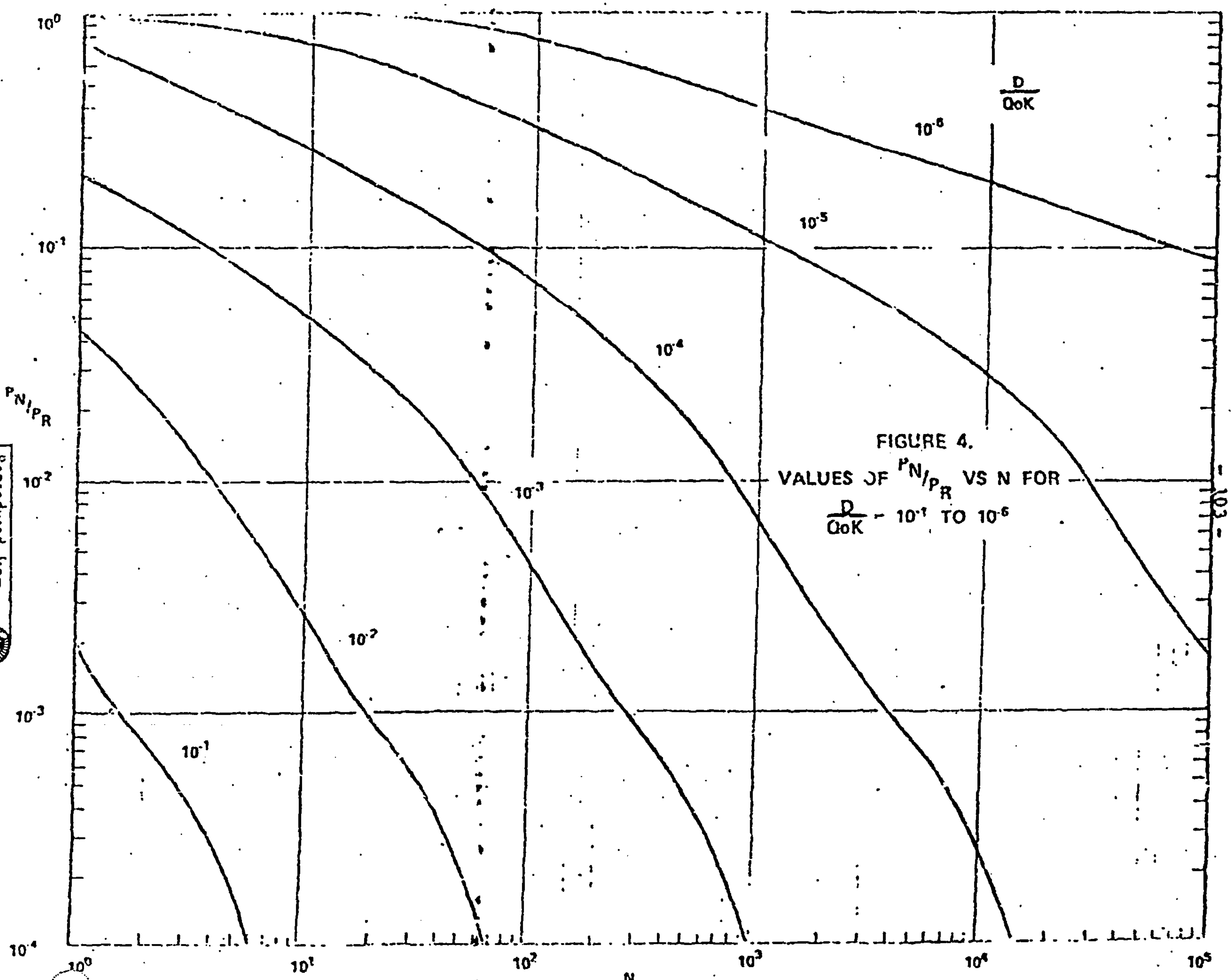

FIGURE 4.
VALUES OF P_N/P_R VS N FOR $\frac{D}{QoK}$ = 10^{-1} TO 10^{-6}

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Appendix "B"

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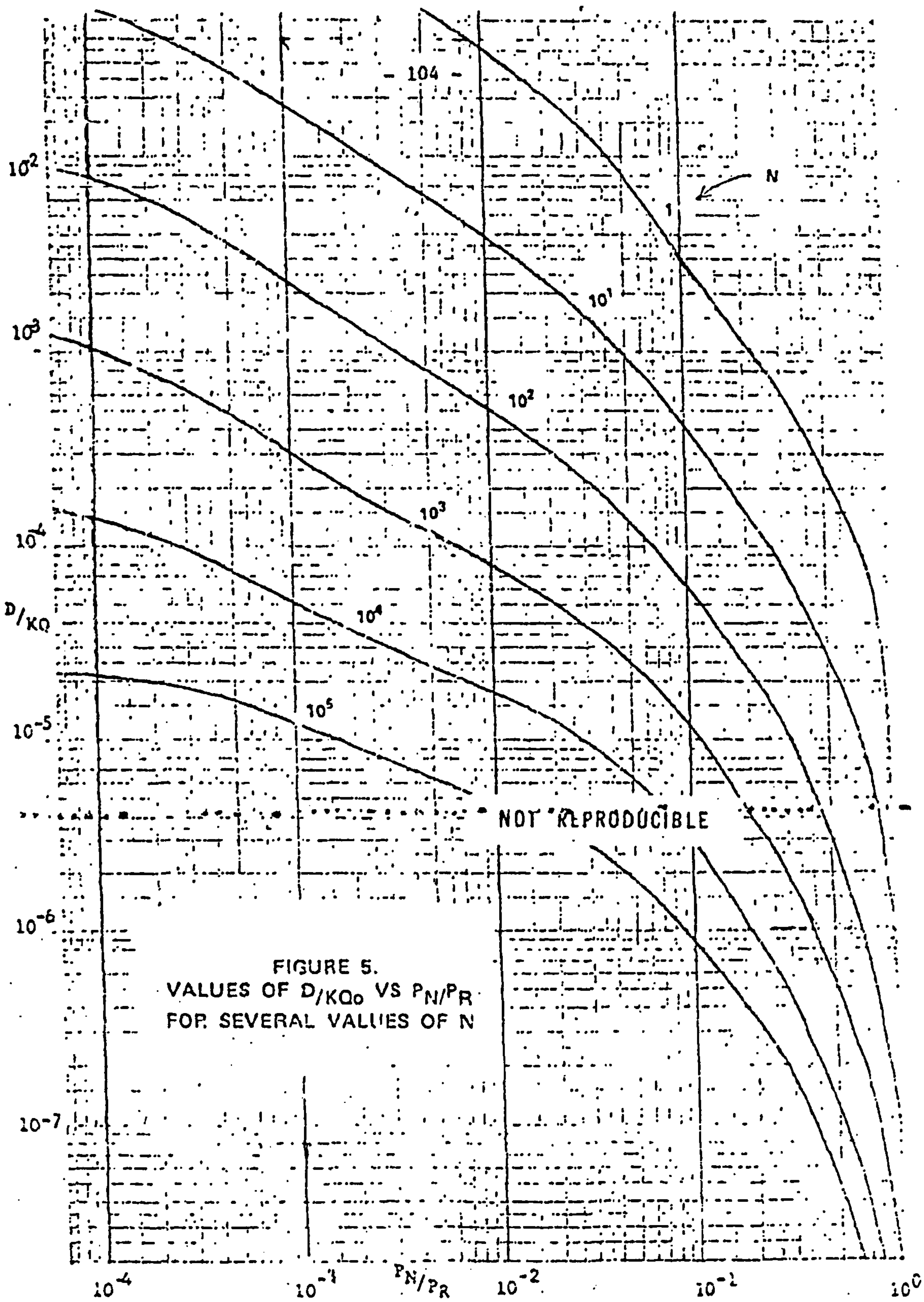


FIGURE 5.
VALUES OF D/KQ_0 VS P_N/P_R
FOR SEVERAL VALUES OF N

Appendix "B"

APPENDIX C

RISKS IN TRANSPORTATION ACCIDENTS FROM COMMON (NONRADIOLOGICAL) CAUSES

Injuries, Fatalities, and Property Damage

In most cases, when a shipment of unirradiated fuel, irradiated fuel, or solid wastes is involved in an accident, the effect on the environment from radiation will be very much less than that from common causes. Statistics supplied by DOT indicate that of the reportable truck accidents in 1969, 33% involved non-fatal injuries and 3.1% involved fatalities. Statistical data on accident probabilities, reportable accidents, and injuries and deaths from common causes are summarized below:

TABLE 1

ACCIDENT STATISTICS - COMMON CAUSES

Mode	Data Year	Probability (Accidents/vehicle-mile)	Injuries Per Accident	Fatalities Per Accident
Truck	1969	1.7×10^{-6}	0.51	0.03
Rail	1969	1.4×10^{-7}*	2.7	0.2
Barge	1970	1.5×10^{-6}	0.06	0.0

*Single rail car.

The following are estimates of the effects from common causes in the shipment of cold fuel to the plant and irradiated fuel and solid waste from the plant and return of both the cold fuel and irradiated fuel shipping containers. If all transport were by truck, the total number of truck miles would be about 155,000 per year. Based on the above data, it is estimated this would cause about 0.1 injuries and 0.01 fatalities per reactor year.

If the cold fuel is transported by truck and the irradiated fuel and solid waste by rail, the total truck miles would be about 12,000 and the total railroad car miles about 15,500 per year. It is estimated this would cause about 0.02 injuries and 0.001 fatalities per reactor year.

If the cold fuel and solid waste are transported by truck and the irradiated fuel by either rail or barge, the total truck miles would be about 35,000, and the total railroad car miles about 10,000 or the total barge miles about 5,000. In either case, it is estimated this would cause about 0.03 injuries and 0.003 fatalities per reactor year.

Also from the 1969 accident statistics for truck transport, about $72 million worth of property damage was reported in about 39,000 accidents or approximately $1800 per accident. The property damage for rail accidents is estimated to average $5800 per accident.[58] Similar data are not available for barge accidents.

The estimated impact on the environment from common causes in transportation associated with the reactor are summarized below:

TABLE 2

Mode of Transport	Environmental Impact for Common Causes - Per Reactor Year		
	Fatalities	Injuries	Property Damage
by truck	0.01	0.1	$475
by truck and rail	0.001	0.02	$ 50
by truck and rail or barge	0.003	0.03	~$120

APPENDIX D

CALCULATIONS OF THE DOSE TO PEOPLE

ALONG THE SHIPPING ROUTE UNDER NORMAL TRANSPORT CONDITIONS

Introduction

This is a description of the method used to calculate the dose to persons along the shipping route from a vehicle containing a shipment of radioactive material. The calculations show that the individual dose to any one person along the route is extremely small and, although large numbers of persons may be receiving this small dose, the cumulative dose to all the persons involved is also small.

The radioactive shipment on the vehicle is a point source for distances from the source of 100 feet or more. For this calculation, based on the regulatory limit of 10 mrem/hr at 6 feet from the surface of the vehicle, the maximum radiation level at 10 feet from the apparent center of the source was estimated to be 10 mrem/hr. The radiation dose to individuals at various distances from the passing source was calculated and summed to determine the total accumulated population dose.

1. The dose rate $\dot{D}$ at an exposure point from a radiation source can be approximated as follows:

$$\dot{D}\ (\mathrm{mrem/hr}) = \frac{K}{r^2}\ e^{-\mu r}\ B(r)$$

where K = constant dependent upon source strength (mrem-ft^2/hr)

r = distance between source and exposure point (ft)

$e^{-\mu r}$ = attenuation factor due to gamma interactions with air occurring between source and exposure point (μ = linear absorption coefficient [1.18×10^{-3} ft^{-1}])

B(r) = buildup factor to account for scattered components returning to exposure point

2. The buildup factor B(r) is difficult to calculate accurately (i.e., with an error less than ~5%) but can be reasonably approximated.[59] The attached graph shows buildup factors as a function of the atomic number Z of the absorbing medium and the distance between the source

and the exposure point for 4 MeV gamma rays. Using values from that graph and assuming B(r) is a linear function, the buildup factor was estimated as follows:

$$\therefore\ B(r) = mr + b$$

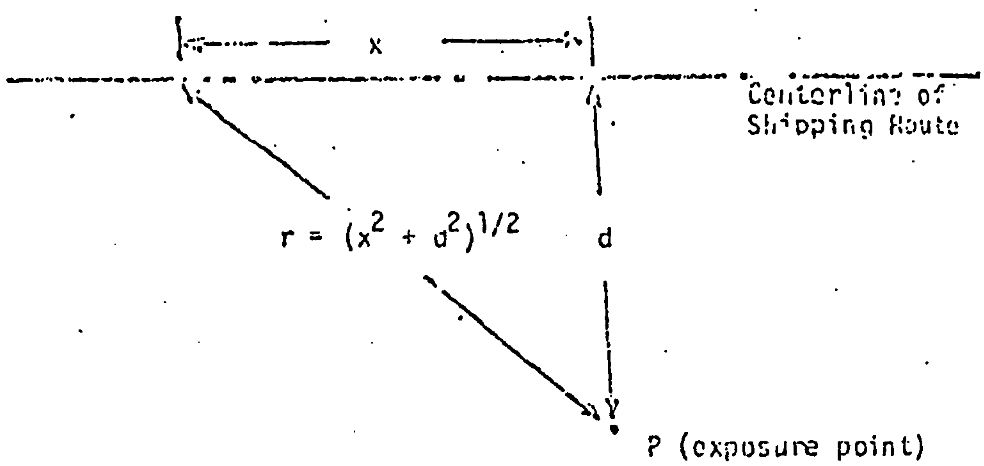
$$\text{at } r = 850 \text{ feet; } B = 1.5$$
$$r = 1700 \text{ feet; } B = 2.0$$

$$\rightarrow m = 6 \times 10^{-4}\ ft^{-1}$$
$$b = 1$$

$$B(r) = (6 \times 10^{-4})r + 1$$

The average gamma ray energy for fission products is known to be about 1 MeV. However, the use of the easily available data for 4 MeV gamma rays will not result in an error which is large compared to the precision of the calculation.

3. The dose to an individual at an exposure point is determined by integrating the dose received by that individual as the radiation source passes his position.



$\dot{D} = \frac{dD}{dt}$ $dD = \dot{D}dt$

x = distance along centerline of shipping route

v = velocity of vehicle = $\frac{dx}{dt}$

d = perpendicular distance from centerline of shipping route

$dD = \frac{1}{v}\dot{D}dx$

total dose $D = \int_{-\infty}^{\infty} \frac{1}{v} \dot{D}dx$
(mrem)

$$D = \frac{K}{v} \int_{-\infty}^{\infty} \frac{e^{-\mu r}}{r^2} B(r)dx$$

$$D(d) = \frac{K}{v} \int_{-\infty}^{\infty} \frac{e^{-\mu (x^2+d^2)^{1/2}}}{(x^2 + d^2)} B([x^2 + d^2]^{1/2})dx$$

Sinc: he integrand is an even function,

$$D(d) = \frac{2K}{v} \int_{0}^{\infty} \frac{e^{-\mu (x^2 + d^2)^{1/2}}}{(x^2 + d^2)} B([x^2 + d^2]^{1/2})dx$$

Since $r^2 = x^2 + d^2$

$2rdr = 2xdx$

and $dx = \frac{r}{x} dr = \frac{r}{(r^2 - d^2)^{1/2}} dr$

$$D(d) = \frac{2K}{v} \int_{d}^{\infty} \frac{e^{-\mu r}}{r^2} B(r) \frac{r}{(r^2 - d^2)^{1/2}} dr$$

$$= \frac{2K}{v} \int_{d}^{\infty} \frac{e^{-\mu r}}{r} \frac{B(r)}{(r^2 - d^2)^{1/2}} dr$$

$$= \frac{2K}{v} \int_{d}^{\infty} \frac{(6 \times 10^{-4} r + 1) e^{-\mu r}}{r(r^2 - d^2)^{1/2}} dr$$

4. In order to obtain a quantitative estimate of dose, the following assumptions were made:

   (a) the source strength, K, is such that the exposure rate is 10 mrem/hr at 10 feet. That is: 10 mrem/hr = $K/10^2$ or

   $K = 10^3$ mrem ft^2/hr.

   (b) the vehicle travels 200 miles/day.

   v (velocity) = 200 miles/day = 200 (5280)/24 (ft/hr)

   $v = 4.4 \times 10^4$ ft/hr

   Based on a uniform distance traveled each day and uniform distribution of persons along the route, the cumulative radiation dose to the population is the same whether the vehicle is moving all of the time at a constant rate of speed or standing still part of the day.

   (c) there are no people closer than 100 feet. As calculated below, the dose to persons farther than 2600 feet from the vehicle is negligible.

   (d) the population density is 330 people/$mile^2$ uniformly dispersed along the route.

Substituting, we have:

$$D(d) = 4.5 \times 10^{-2} \int_d^{\infty} \frac{[6 \times 10^{-4} r + 1]}{r} \frac{e^{-\mu r}}{[r^2 - d^2]^{1/2}} dr$$

D is the total dose (mrem) a person standing a distance d from the centerline of the shipping route would receive from the passing vehicle.

Integrating the above expression numerically yields the values given in Table I.

TABLE I

Distance from Centerline of Shipping Route (feet)	Individual Dose at Given Distance (mrem)
100	5.8×10^{-4}
200	2.5×10^{-4}
300	1.5×10^{-4}
400	1×10^{-4}
500	7.1×10^{-5}
700	4×10^{-5}
900	2.5×10^{-5}
1000	2×10^{-5}
1300	1.1×10^{-5}
1500	7.8×10^{-6}
1700	5.5×10^{-6}
2000	3.4×10^{-6}
2300	2.1×10^{-6}
2600	1.3×10^{-6}

Note:
Doses at some intermediate distances have been omitted to shorten the table.

5. In order to obtain the man-rem dose, it was assumed that on the average in each mile of the shipping route, a total of 165 people are uniformly distributed between 100 feet and 2600 feet on each side of the route. For ease of calculation, 1/26th of the 165 people are considered to be grouped at 100 feet intervals on each side of the route.

The total man-rem dose per vehicle mile to the persons on one side of the route is:

$$(165/26) \text{ people/mile } (D(100 \text{ ft}) + D(200 \text{ ft}) + \ldots + D(2600 \text{ ft}))$$

$$= (165/26) \text{ people/mile } (5.8 \times 10^{-7} \text{ rem} + 2.5 \times 10^{-7} \text{ rem} + \ldots + 1.3 \times 10^{-9} \text{ rem})$$

$$= 6.35 \, (1.4 \times 10^{-6}) \text{ man-rem/mile}$$

$$= 9.0 \times 10^{-6} \text{ man-rem/mile}$$

For both sides of the route, the cumulative dose is about 1.8×10^{-5} man-rem/mile.

For example, if the source travels 1000 miles, the total cumulative dose would be 2×10^{-2} man-rem. The total dose to the individual receiving the most exposure under the conditions assumed from a single shipment would be about 6×10^{-4} mrem.

The average population density in most cases is assumed to be 330 persons per square mile. This represents an area in which the population density is high, such as along the East Coast. For the area west of the Mississippi other than California, an average population density of 110 persons per square mile should be used as being more representative of that region. For shipment by barge, it is estimated that for the average barge route no persons reside within half a mile on either side of 2/3 of the route.

6. <u>Conclusions</u>

The cumulative dose to persons along the route of shipments of unirradiated and irradiated fuel and solid wastes, based on the shipment traveling 200 miles per day, estimated radiation levels in the vicinity of the transporting vehicle shown below and population densities discussed above, the population dose in man-rem for each mile over which unirradiated fuel, irradiated fuel or solid waste is shipped is given in Table II.

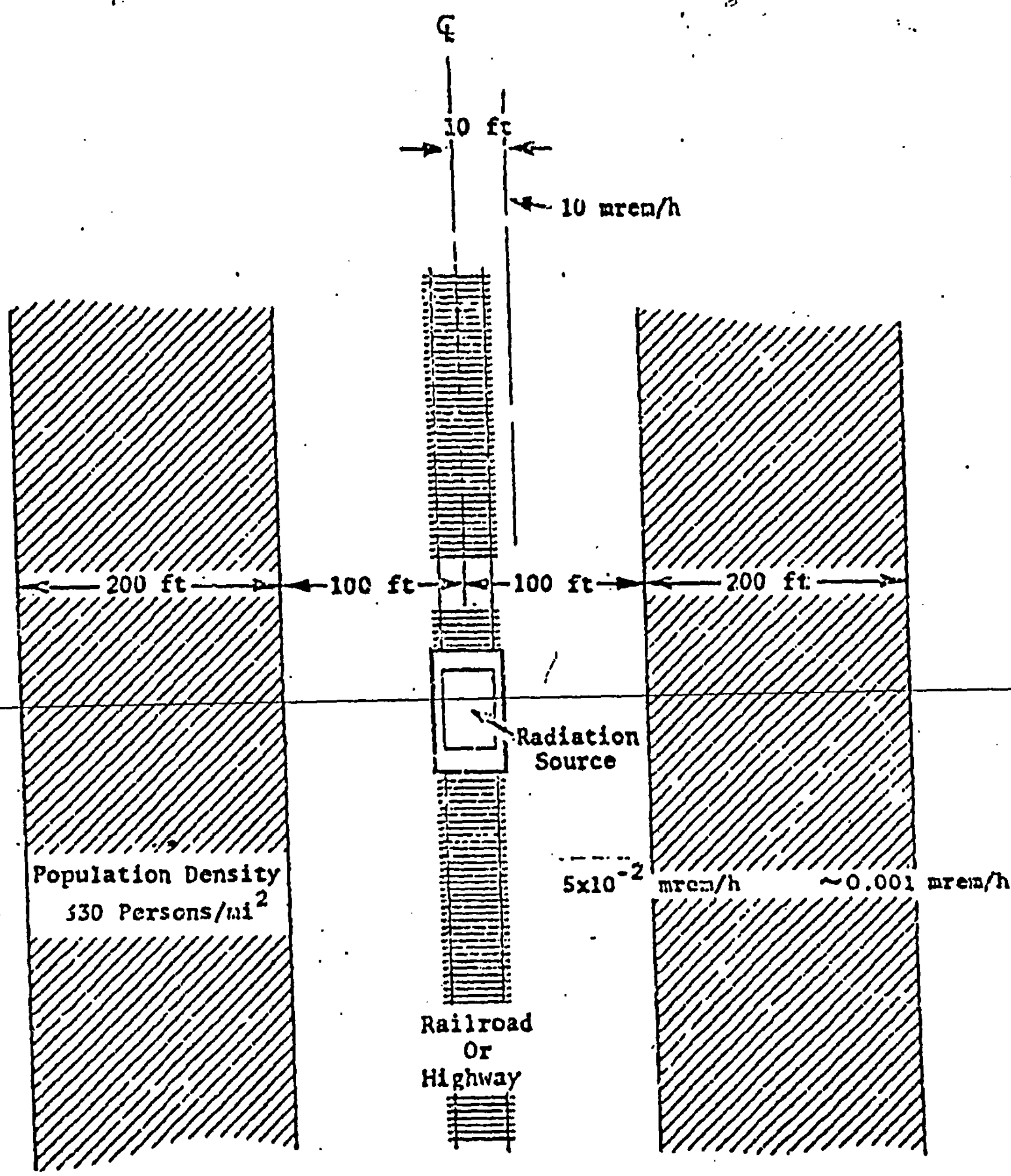
TABLE II

POPULATION DOSE PER MILE
SHIPMENT TRAVELS

Type of Shipment	Mode of Transport	Estimated(1) Radiation Level (mrem/hr)	Number(2) of Persons Exposed	Cumulative Population Dose per Mile (man-rem)
Unirradiated nuclear fuel	Truck	0.1	300	1.8×10^{-7}
Irradiated fuel	Truck	10	300	1.8×10^{-5}
	Rail		300	1.8×10^{-5}
	Barge		100	6×10^{-6}
Solid radio-active waste	Truck	10	300	1.8×10^{-5}
	Rail		300	1.8×10^{-5}

(1) Radiation level estimated at 10 feet from apparent center of source.

(2) Average number of persons within 1/2 mile of centerline of route.



Not To Scale

FIGURE 1. POPULATION DISTRIBUTION ALONG SHIPPING ROUTE

Appendix "D"

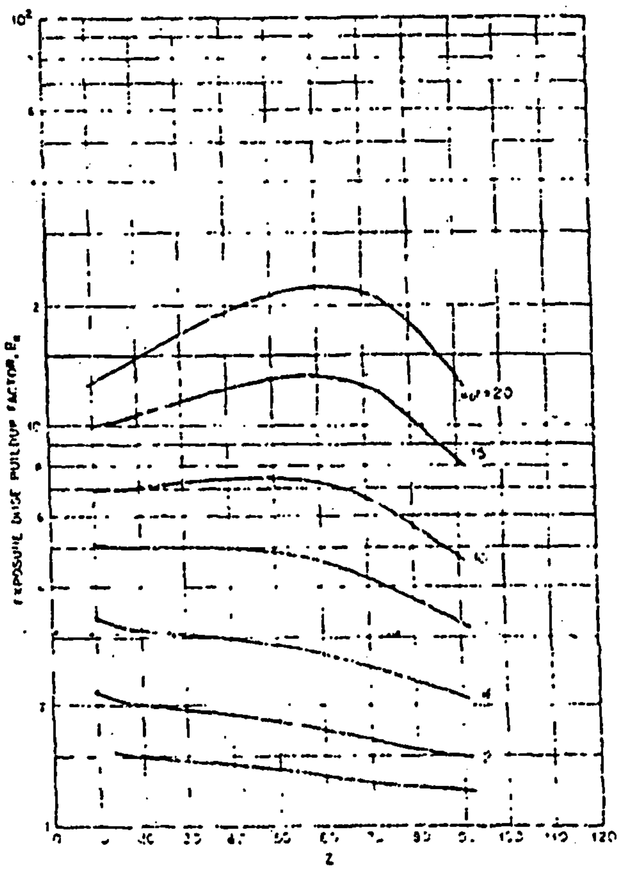


Fig. 5.45—Exposure-dose buildup factor as a function of atomic number *Z*, for a 4-MeV point isotropic source. (From H. Goldstein, *Fundamental Aspects of Reactor Shielding*, Addison-Wesley Publishing Co., Reading, Mass., 1959.)

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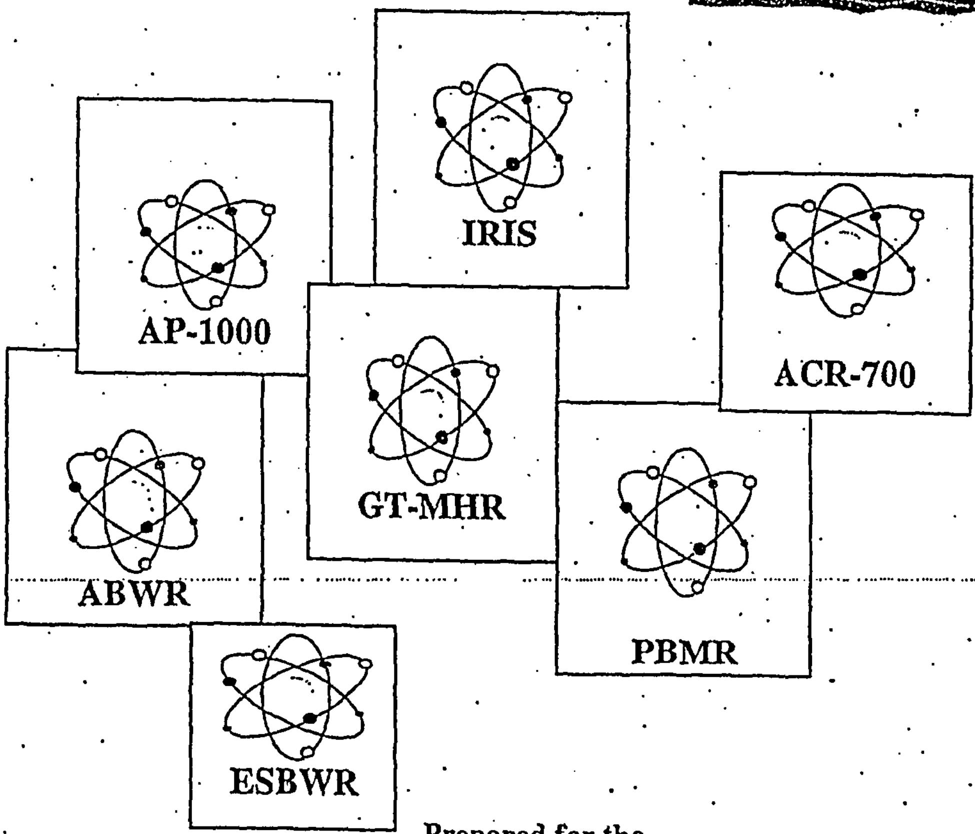
Early Site Permit Environmental Report Sections and Supporting Documentation



Prepared for the

United States Department of Energy
Office of Nuclear Energy, Science and Technology

May 15, 2003

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431.02
01/30/2003
Rev. 11

ENGINEERING DESIGN FILE

EDF No.: 3747 EDF Rev. No.: 0 Project File No.: N/A

1. Title: Early Site Permit ER Sections and Supporting Documentation

2. Index Codes:
Building/Type N/A SSC ID N/A Site Area N/A

3. NPH Performance Category: ______ or ☒ N/A

4. EDF Safety Category: ______ or ☒ N/A SCC Safety Category: ______ or ☒ N/A

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ENGINEERING DESIGN FILE

EDF No.: 3747 EDF Rev. No.: 0 Project File No.: N/A

1. Title: Early Site Permit ER Sections and Supporting Documentation

2. Index Codes:
   Building/Type N/A SSC ID N/A Site Area N/A

13. Registered Professional Engineer's Stamp (if required)

EDF-3747
Early Site Permit ER Sections and Supporting Documentation
Table of Contents

ER SECTION 3.8
TRANSPORTATION OF RADIOACTIVE MATERIALS

3.8 Transportation of Radioactive Materials

This section addresses the transportation issues associated with siting and operating a new reactor and is divided into two main subsections. The first subsection addresses the light-water-cooled reactor (LWR) designs presently being considered. The second subsection addresses the gas-cooled reactor designs also being considered. This split addresses the regulatory distinction made in 10 CFR 51.52 for light-water-cooled reactors.

3.8.1 Light-Water-cooled Reactors

As required by 10 CFR 51.52, every environmental report prepared for the construction permit stage of a light-water-cooled nuclear power reactor (LWR), and submitted on or after September 4, 1979, is to utilize Table S-4, "Environmental Impact of Transportation of Fuel and Waste To and From One Light-Water-Cooled Nuclear Power Reactor," and shall contain a statement concerning transportation of fuel and radioactive wastes to and from the reactor.

Table S-4 (as provided in 10 CFR 51.52(c) and repeated in Table 3.8-3) is a summary impact statement concerning transportation of fuel and radioactive wastes to and from a reactor. The table is divided into two categories of environmental considerations: (1) normal conditions of transport and (2) accidents in transport. The normal conditions of transport consideration are further divided into environmental impact, exposed population, and range of doses to exposed individuals per reactor reference year. The "accidents in transport" consideration is concerned with environmental risk. Under "normal conditions of transport," the environmental impacts of the heat of the fuel cask in transit, weight, and traffic density are described. Also the number and range of radioactive doses to transportation workers and the general public are described. Under "accidents in transport," the environmental risk from radiological effects and common nonradiological causes such as fatal and nonfatal injuries and property damage are described.

To indicate that Table S-4 adequately describes the environmental effects of the transportation of fuel and waste to and from the reactor, the reactor licensee must state that the reactor and this transportation either meet all of the conditions in paragraph (a) of 10 CFR 51.52 or all of the conditions in paragraph (b) of 10 CFR 51.52. Subparagraphs 10 CFR 51.52(a)(1) through (5) delineate specific conditions the reactor must meet to use Table S-4 as part of its environmental report. Subparagraph 10 CFR 51.52(a)(6) states, "The environmental impacts of transportation of fuel and waste to and from the reactor, with respect to normal conditions of transport and possible accidents in transport, are as set forth in Summary Table S-4 in paragraph (c) of this section; and the values in the table represent the contribution of the transportation to the environmental costs of licensing the reactor." Paragraph 10 CFR 51.52(b) states that reactors not meeting the conditions of 10 CFR 51.52(a) shall make a full description and detailed analysis for their reactor equivalent to Table S-4.

The light water cooled reactor technologies being considered have characteristics that fall within the conditions of 10 CFR 51.52, for use of Table S-4, with one minor exception for two of the reactor designs, i.e., rated core thermal power level. The effect of this difference will be discussed later.

The light water cooled technologies being considered are identified in Section 1.1.3. These designs include the ABWR (Advanced Boiling Water Reactor), the ESBWR (Economic Simplified Boiling Water Reactor), the AP-1000 (Advanced Passive PWR), the IRIS (International Reactor Innovative and Secure), and the ACR-700 (Advanced CANDU Reactor). The standard configuration for each of these reactor technologies is as follows. The ABWR is a single unit, 4300 MWt, 1500 MWe reactor. The ESBWR is a similar BWR: single unit, 4000 MWt, 1390 MWe. The AP-1000 is a single unit, 3400 MWt, 1117-1150 MWe pressurized water reactor. The IRIS is a three module pressurized water reactor configuration for a total of 3000 MWt and 1005 MWe. And the ACR-700 is a twin unit, 3964 MWt, 1462 MWe, light-water-cooled reactor with a heavy water moderator.

10 CFR 51.52 lists several conditions that need to be addressed by these reactor technologies. If all the conditions are satisfied by all of the reactor technologies, then the Table S-4 values are appropriate for use in the Early Site Permit. These conditions are reactor core thermal power; fuel form; fuel enrichment; fuel encapsulation; average fuel irradiation; time after discharge of irradiated fuel before shipment; mode of transport for unirradiated fuel; mode of transport for irradiated fuel; and mode of transport for radioactive waste other than irradiated fuel. There are two other conditions in S-4 that require that all radioactive waste, with the exception of irradiated fuel, be packaged and in solid form. Table 3.8-1, "LWR Transportation Worksheet," was prepared to succinctly show the reference conditions along with the values for the new reactor technologies. The information to complete the table was supplied by the reactor vendors.

10 CFR 51.52(a)(1) requires that the reactor have a core thermal power level not exceeding 3800 megawatts. Of the considered LWR technologies, only the two boiling water reactors, the ABWR and the ESBWR, exceed this value. The ABWR has a core thermal power level of 4300 megawatts thermal (MWt) while the ESBWR reactor power level is 4000 MWt. The higher rated core power level would typically indicate the need for more fuel and therefore more fuel shipments. This is not the case in this instance due to the higher unit capacity and higher burnup for the reactors with the increased power level. The annual fuel loading for the reference reactor was 35 MTU while the annual fuel loading for both the ABWR and ESBWR is only 32.8 MTU. In fact, the annual MTU of fuel normalized to equivalent electrical generation is just slightly more than half of the reference LWR, 18.4 versus 35. This reduced annual MTU of fuel will mean fewer shipments and less environmental impact. Also, WASH-1238 states: "The analysis is based on shipments of fresh fuel to and irradiated fuel and solid waste from a boiling water reactor or a pressurized water reactor with design ratings of 3,000 to 5,000 megawatts thermal (MWt) or 1,000 to 1,500 megawatts electrical (MWe)." Both the ABWR and the ESBWR fall within these bounds.

10 CFR 51.52(a)(2) requires that the reactor fuel be in the form of sintered uranium dioxide (UO_2) pellets. The LWR technologies being considered have a sintered UO_2 pellet fuel form.

10 CFR 51.52(a)(2) requires that the reactor fuel have a uranium-235 enrichment not exceeding 4% by weight. This condition has been modified by "NRC Assessment of the Environmental Effects of Transportation Resulting From Extended Fuel Enrichment and Irradiation" as provided in 53FR30555 and 53FR32322. This reference along with NUREG 1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, concluded that 5% enrichment is also bounded. Based on this modification, the LWR technologies being considered meet this condition. NOT Applc.

10 CFR 51.52(a)(2) requires that the reactor fuel pellets be encapsulated in Zircaloy rods. This has been modified by 10 CFR 50.44, which allows use of ZIRLO™. Based on this modification, the LWR technologies being considered meet this condition.

10 CFR 51.52(a)(3) requires that the average burnup is not to exceed 33,000 megawatt-days per metric ton of uranium (MWd/MTU). NUREG 1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, concludes that 62,000 MWd/MTU for the peak rod is also bounded by the Table. Based on this modification, the LWR technologies being considered meet this condition. The average discharge burnup in MWd/MTU ranges from a low of 20,500 for the ACR-700 to a high of 55,200 for the IRIS reactor technology.

10 CFR 51.52(a)(3) requires that no irradiated fuel assemblies be shipped until at least 90 days after it is discharged from the reactor. Table S-4 assumes 150 days of decay time prior to shipment of any irradiated fuel assemblies. For the LWR technologies being considered, five years is the minimum decay time expected before shipment of irradiated fuel assemblies. The five-year minimum time is supported additionally by two current practices. One is per contract with DOE, who has ultimate responsibility for the spent fuel. Five years is the minimum cooling time specified in 10 CFR 961, Appendix E. The other practice is the NRC specifies five years as the minimum cooling period when they issue certificates of compliance for casks used for shipment of power reactor fuel. (NUREG-1437, Addendum 1, pp 26) In all likelihood, the decay time will be at least ten years and probably even longer. In addition to the minimum fuel storage time, NUREG-1555 Environmental Standard Review Plan, Section 3.8 asks for the capacity of the onsite storage facilities to store irradiated fuel. The LWR technologies being considered are designing for on-site storage of spent fuel for up to 60 years through a combination of pool and dry storage.

10 CFR 51.52(a)(5) requires that unirradiated fuel be shipped to the reactor by truck. The LWR technologies being considered are planning to ship their unirradiated fuel by truck.

10 CFR 51.52(a)(5) allows for truck, rail, or barge transport of irradiated fuel. The LWR technologies being considered comply with the transport mode. Three of the reactor vendors identified rail as the shipment mode, two reactor vendors specified truck as the

shipment mode, and the vendor for the ABWR and the ESBWR stated either rail or truck. Of note, the DOE is responsible for transport from reactor sites to the repository and DOE will make the decision on transport mode. NUREG-1555, Environmental Standard Review Plan, Section 3.8, also asks for the estimated transportation distance from the plant to the facility to which irradiated fuel will most likely to be sent. Recognizing the uncertainty in predicting the future destination of spent fuel in the United States, 2500 miles is utilized as a bounding distance at this time. This length bounds the approximate average distance from typical reactor sites to potential repository locations in the US.

10 CFR 51.52(a)(5) requires that the mode of transport of low-level radioactive waste is either truck or rail. The LWR technologies being considered plan to ship their radioactive waste by truck.

Finally, 10 CFR 51.52(a)(4) requires that with the exception of spent fuel, radioactive waste shipped from the reactor is to be packaged and in a solid form. The LWR technologies being considered will solidify and package their radioactive waste. Additionally, existing NRC (10 CFR 71) and DOT (49 CFR 173,178) packaging and transportation regulations specify requirements for the shipment of radioactive material. The LWR technologies being considered are also subject to these regulations.

In conclusion, since the LWR technologies being considered satisfy the basis 10 CFR 51.52(a) conditions for use of Table S-4, the environmental impacts of transportation of fuel and radioactive wastes are represented by the values given in 10 CFR 51.52(c), Table S-4. Thus, the radiological and nonradiological environmental impacts of transportation of fuel to and from, and waste from, an LWR are small.

Table 3.8-1 LWR-S4 Transportation Worksheet

Reactor Technology Characteristic	Table S-4 Condition	ESBWR (Single unit) (4000 MWt) (1390 MWe)	ABWR (Single unit) (4300 MWt) (1500 MWe)	AP-1000 (Single Unit) (3400 MWt) (1117 - 1150 MWe)	IRIS (3 Reactors) (3000 MWt total) (1005 MWe total)	ACR-700 (Twin Unit) (3964 MWt total) (1462 Mwe total)
Reactor Power Level MWt	not exceeding 3800 MWt per reactor	4000 MWt	4300 MWt	3400	3000 (1000 MWt per reactor, 3 reactors per plant)	3964 (1982 MWt per reactor, 2 reactors per plant)
Fuel Form	sintered UO_2 pellets	sintered UO_2 pellets	sintered UO_2 pellets	sintered UO_2 pellets	sintered UO_2 pellets	sintered UO_2 pellets
U235 Enrichment	Not exceeding 4%; NUREG 1437 concludes that 5% is bounded	Initial Core < 3.5%; Reload average < 4.5%	Initial Core < 3.5%; Reload average < 4.5%	Initial Core Load Region 1 2.35% Region 2 3.40% Region 3 4.45% Reload Average 4.51%	fuel cycle average ~ 4.85%; maximum assembly 4.95%; reload 4.75 - 4.95%	2%
Fuel Rod Cladding	Zircaloy rods; 10 CFR 50.44 allows use of ZIRLO	Zircaloy	Zircaloy	Zircaloy or ZIRLO™	ZIRLO™	Zircaloy-4
Average burnup MWd/MTU	Not exceeding 33,000; NUREG 1437 concludes 62,000 MWd/MTU for peak rod is bounded	46,000	46,000	48,700	55,200	20,500

LWR-S4 Transportation Worksheet

Reactor Technology	Reference LWR (Single unit) (1100 MWe)	ESBWR (Single unit) (4000 MWt) (1390 MWe)	ABWR (Single unit) (4300 MWt) (1500 MWe)	AP-1000 (Single Unit) (3400 MWt) (1117 - 1150 MWe)	IRIS (3 Reactors) (3000 MWt total) (1005 MWe total)	ACR-700 (Twin Unit) (3964 MWt total) (1462 Mwe total)
Characteristic						
Unirradiated fuel						
transport mode	truck	truck	truck	truck	truck	truck
Irradiated fuel						
transport mode	truck, rail or barge	truck, rail	truck, rail	rail	rail	rail
decay time prior to shipment	Not less than 90 days is a condition for use of Table S-4; 5 years is per contract with DOE	five years	five years	ten years	five years	ten years
radioactive waste						
transport mode	truck or rail	truck	truck	truck	truck	truck
waste form	solid	solid	solid	solid	solid	solid
packaged	yes	yes	yes	yes	yes	yes

Yellow indicates a value larger than or different from Table S-4

3.8.2 Gas-cooled Reactors

3.8.2.1 Introduction and Background

The following assessment of the environmental impacts of the transportation of fresh and spent fuel and low-level waste to and from the reactor for gas-cooled reactor technologies is based on a comparison of the key parameters and conditions that were used to generate the impacts listed in 10 CFR 51.52(c), Table S-4. This comparison can then demonstrate that the environmental impacts of these gas-cooled reactor technologies are no worse than the impacts previously identified in Table S-4 for the light-water-cooled technologies. The premise being that if the values of the major contributors to the health and environmental impacts that were used for the reference LWR are greater than those comparable values for the gas-cooled reactor technologies, then the subsequent impacts would also be greater and therefore bounding. It is important to point out that even though we are looking at the contributors individually, it is the overall cumulative impact that is of concern. That is, for purposes of comparing/evaluating cumulative impacts, there can be increases in select individual contributors if offset by decreases in other contributors.

The parameters that have been chosen for purposes of comparison include not only the major contributors to the health and environmental impacts but also the conditions listed in 10 CFR 51.52. The major contributor to transportation risk is the number of shipments. Basically, the more shipments, the more risk; if there are no shipments, there is no risk. The Table S-4 shipments include fresh fuel for both initial core loading and reloads, irradiated fuel, and low-level waste (LLW) from operations. The second main contributor to the transportation risk would be the mode of shipment. In this case, only trucks and trains are considered. The last important risk factor relates to what kind of material is being shipped. In the category for irradiated fuel, we compared fission product inventory, krypton inventory, actinide inventory, total radioactivity, decay heat, and weight of shipment. For radioactive waste, we used the volume to determine the number of shipments. Radioactivity (Ci) was also estimated to assure that the assumption about the percentage of LLW that might require shielding was reasonable.

The 10 CFR 51.52 conditions are: reactor core thermal power; fuel form; fuel enrichment; fuel encapsulation; average fuel irradiation; time after discharge of irradiated fuel before shipment; mode of transport for unirradiated fuel; mode of transport for irradiated fuel; and mode of transport for radioactive waste other than irradiated fuel. In addition, there are two other conditions that require that all radioactive waste with the exception of irradiated fuel be packaged and in solid form. Since existing packaging and transportation regulations already address those items and would also apply to these new reactor technologies, no further discussion is needed for these two conditions.

Before proceeding with the evaluation, it is important to note that the NRC has an ongoing review of the safety of spent fuel transportation. The latest evaluation is

NUREG/CR-6672, "Reexamination of Spent Fuel Shipment Risk Estimates," published in March 2000. The NRC in their document "An Updated View of Spent Fuel Transportation Risk," concluded that the NUREG/CR-6672 study confirmed that earlier risk estimates (NUREG-0170, "Final Environmental Statement on the Transport of Radioactive Materials by Air and Other Modes") to the public remain conservative by factors of 2 to 10 or more; that existing regulations governing the shipment of spent fuel are adequate; and no unreasonable risk is posed to the public by the continued shipment of spent fuel. The range of conservative risk factors covers differences in mode of transport (rail or truck) and either accident or accident-free scenarios. 2

These same NRC conclusions support the position that environmental assessments of the transport casks do not have to be done for the Part 71 cask certifications because they meet the categorical exclusion criteria in 10 CFR 51.22(c)(13) that package designs used for the transportation of licensed materials do not require an environmental review. As discussed in 10 CFR 51.22(a), the NRC has determined that certain categories of licensing and regulatory actions have already been determined individually or cumulatively to not have a significant effect on the human environment; thus, a separate environmental assessment is not required. As mentioned in the previous paragraph, a generic assessment of the environmental effects associated with transportation of all radioactive material, including spent fuel, has already been done as provided in NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," dated December 1977. This environmental impact statement (EIS) provided the regulatory basis for continued issuance of general licenses for transportation of radioactive material under 10 CFR 71. In addition, the NRC has conducted a reexamination of the risks associated with spent fuel shipments as documented in NUREG/CR-6672. This reexamination concluded that the estimated risks for future shipments are well below those in the 1977 study. Thus, NUREG-0170 remains valid as the baseline report on which National Environmental Policy Act (NEPA) analyses of transportation risk are based.

Table 3.8-1 captures the major features of the reference LWR that were used to develop Table S-4 and compares these same features with the gas-cooled reactor technologies being considered. The reference LWR pertains to the typical 1100 MWe light-water-cooled nuclear reactor as described in WASH-1238. The information to construct the worksheet was taken from the "Normal Conditions of Transport" portion of the 10 CFR 51.52 Summary Table S-4 "Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor," WASH-1238 "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants" and Supplement 1 to WASH-1238 (NUREG-75/038) for the reference LWR. The information for the reactor technologies was provided by the reactor vendors.

3.8.2.2 Analysis

This section provides a detailed description of the comparison of the individual characteristics supporting Table S-4 against the corresponding parameters for the gas-cooled reactor technologies. The value for the reference reactor is given along with the corresponding values or range of values for the gas-cooled reactor technologies. As appropriate, additional information and/or observations are provided. Table 3.8-2, the Gas-cooled Reactor Transportation Worksheet, provides additional details regarding the reactor technology specific values.

There are two gas-cooled reactor technologies presently being considered. These reactor technologies are the GT-MHR (Gas Turbine-Modular Helium Reactor), and the PBMR (Pebble Bed Modular Reactor). The standard configuration for each of these reactor technologies is as follows. The GT-MHR is a four module, 2400 MWt, 1140 MWe gas-cooled reactor. The PBMR is an eight module, 3200 MWt, 1320 MWe gas-cooled reactor. The unit capacities for these reactors are as follows: 88% for the GT-MHR; 95% for the PBMR. These values are contrasted with the reference LWR, a single unit, 1100 MWe plant with a unit capacity factor of 80%.

1320
.95
1254
880/1254 = 0.70 (-30%)

1140 MWe
.88
1003 MWe
880/1003 = 0.88 (-12%)

Before beginning direct comparisons, it is important to note that the plants being considered are a different physical size, have a different electrical rating, and have a different capacity factor from the reference LWR. In order to make proper comparisons, we need to evaluate the characteristics based on equivalent criteria. In this case, electrical generation is the metric of choice. Electrical generation is why the plants are being built, and we want to know if these new reactor technologies, for the same electrical output, have a greater or lesser impact on the health and environment. The reference LWR is an 1100 MWe plant with a capacity factor of 80%. Based on this, the reactor technologies should be normalized to 880 MWe using their plant specific electrical rating and capacity factor. For many of the characteristics being examined, this adjustment is not necessary. But in a few cases, specifically those dealing with the number of shipments of fuel and waste, an adjustment is appropriate. The amount of this adjustment ranges from minus 12% for the GT-MHR to minus 30% for the PBMR.

$1-\left[880/(1140\times0.88)\right]$ $1-\left[880/(1320\times0.95)\right]$

3.8.2.3 Table S-4 Conditions

As discussed previously, Table S-4 lists several conditions that need to be addressed by the new reactor technologies. These conditions are reactor core thermal power; fuel form; fuel enrichment; fuel encapsulation; average fuel irradiation; time after discharge of irradiated fuel before shipment; mode of transport for unirradiated fuel; mode of transport for irradiated fuel; and mode of transport for radioactive waste other than irradiated fuel. Two other conditions in S-4 require that radioactive waste, with the exception of irradiated fuel, be packaged and in solid form.

10 CFR 51.52(a)(1) requires that the reactor have a core thermal power level not exceeding 3800 MWt. The gas-cooled reactors being considered meet this condition. The GT-MHR has a core thermal power level of 600 MWt per module. The PBMR has a core thermal power level of 400 MWt per module.

10 CFR 51.52(a)(1) requires that the reactor fuel be in the form of sintered UO_2 pellets. The fuel form for the gas-cooled reactors being considered is TRISO coated uranium

oxycarbide fuel kernels for the GT-MHR and TRISO coated uranium dioxide fuel kernels for the PBMR.

10 CFR 51.52(a)(2) requires that the reactor fuel have a uranium-235 enrichment not exceeding 4% by weight. This has been modified by NUREG 1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, which concluded that 5% enrichment is also bounded. The PBMR has an equilibrium enrichment of 12.9% while the GT-MHR fissile particle enrichment is 19.8%.

10 CFR 51.52(a)(2) requires that the reactor fuel pellets be encapsulated in Zircaloy rods. This has been modified by 10 CFR 50.44, which allows use of ZIRLO. The gas-cooled reactors being considered have a different configuration. The fuel kernels are coated with layers of pyrolytic carbon and silicone carbide. These coatings are considered the equivalent of the fuel cladding. For the GT-MHR these TRISO fuel particles are blended and bonded together with a carbonaceous binder. These are stacked within a graphite block. For the PBMR, the fuel unit is a 6 cm diameter graphite sphere containing approximately 15000 TRISO fuel particles.

10 CFR 51.52(a)(3) requires that the average burnup is not to exceed 33,000 MWd/MTU. NUREG 1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, concludes that 62,000 MWd/MTU for the peak rod is also bounded by the Table. The gas-cooled reactors have an expected burnup of 133,000 MWd/MTU for the PBMR and 112,742 MWd/MTU for the GT-MHR.

10 CFR 51.52(a)(3) requires that no irradiated fuel assemblies be shipped until at least 90 days after it is discharged from the reactor. Table S-4 assumes 150 days of decay time prior to shipment of any irradiated fuel assemblies with a condition of not less than 90 days. For the gas-cooled reactor technologies being considered, five years is the minimum decay time prior to shipment of irradiated fuel assemblies. This is per contract with DOE, who has ultimate responsibility for the spent fuel. In all likelihood, the decay time will be at least ten years and probably even longer. The gas-cooled reactor technologies being considered are designing for on-site storage of spent fuel for up to 60 years including potential modular storage expansions.

10 CFR 51.52(a)(3) requires that the unirradiated fuel be shipped to the reactor by truck. The gas-cooled reactor technologies being considered are planning to ship their unirradiated fuel by truck.

10 CFR 51.52(a)(3) allows for truck, rail, or barge transport of irradiated fuel. The gas-cooled reactor technologies being considered plan to allow for irradiated fuel shipment by truck. However, the actual mode of shipment will be determined by DOE and may include either rail or truck shipments.

10 CFR 51.52(a)(3) requires that the mode of transport of low-level radioactive waste is either truck or rail. The gas-cooled reactor technologies being considered plan to ship their radioactive waste by truck.

Finally, 10 CFR 51.52(a)(4) requires that that, with the exception of spent fuel, radioactive waste shipped from the reactor is to be packaged and in a solid form. The gas-cooled technologies being considered will solidify and package their radioactive waste. Additionally, existing NRC (10 CFR 71) and DOT (49 CFR 173,178) packaging and transportation regulations specify requirements for the shipment of radioactive material. The gas-cooled technologies being considered are also subject to these regulations.

3.8.2.4 Risk Contributors – Shipments

This section discusses the type and number of shipments for the gas-cooled reactor technologies and the values used for the reference LWR.

The reference LWR assumed an initial core loading of 100 MTU for a PWR and 150 MTU for a BWR. These quantities resulted in 18 truck shipments. For the new gas-cooled reactor technologies, the numbers of shipments were 44 for the PBMR and 51 for the GT-MHR. If normalized to the equivalent electrical output, the number of shipments would be 31 and 45 respectively.

51 • 0.88

4 and 5

The reference LWR assumed an annual reload of 30 MTU. This quantity resulted in 6 truck shipments. For the new gas-cooled reactor technologies, the numbers of reload shipments ranged from 2 for the PBMR to 20 for the GT-MHR.. The number of shipments normalized to the electrical generation changes slightly to 18 for the GT-MHR.

/unit

PBMR not changed to 20

I got 26/yr for PBMR

With respect to the number of spent fuel shipments by truck, the reference LWR assumed 60 shipments annually. For the two gas-cooled reactor technologies, the number of shipments is considerably less. The PBMR requires 16 annual shipments while the GT-MHR requires 38 truck shipments annually. Normalizing to the electrical generation lowers these numbers to 12 to 34, respectively.

The reference LWR assumed 10 rail shipments annually of spent fuel. Since the gas-cooled reactor technologies are not planning to ship their spent fuel by rail, no comparison is needed. However, based on the comparison for truck shipments, fewer than 10 rail shipments annually would be expected if DOE decided to use larger and higher capacity rail transport casks for gas-reactor spent fuel.

Larger than what? current rail & truck casks

The reference LWR also considered transporting spent fuel by barge and assumed 5 shipments annually. Since the gas-cooled reactor technologies are not planning to ship their spent fuel by barge, no comparison is needed.

The reference LWR assumes 46 shipments annually of low-level radioactive waste. The gas-cooled reactor technologies will make far fewer shipments. The GT-MHR will need only 6 shipments while the PBMR will require 9 shipments annually. These results assume that 90% of the LLW can be shipped at 1000 ft^3 per truck, and the remaining 10%

can be shipped at 200 ft^3 per truck. If the numbers are normalized to electrical generation, the numbers of shipments range from 6 to 7.

The Table S-4 value, traffic density in trucks per day, for the reference LWR is given as less than one per day. Both the gas-cooled reactor technologies would also have less than one per day. In fact, the new gas-cooled reactor technologies would have far fewer shipments per year. The reference LWR bounding annual value for truck shipments is 110 based on a 40 year period, while the normalized number of truck shipments for the gas-cooled reactor technologies would require as few as 18 for the PBMR and only 41 for the GT-MHR.

The rail density in cars per month for the reference LWR is given as less than 3 per month. Since the gas-cooled reactor technologies are not planning to make any shipments by rail, no comparison is needed. However, as noted above, if DOE decided to use rail transport for spent fuel instead of truck, fewer than 3 shipments per month would be expected based on the expected larger capacity of rail spent fuel casks compared to truck casks.

3.8.2.5 Risk Contributors - Contents

This section addresses the radioactive contents of the shipments and their thermal loading and compares them to the reference LWR. The radioactive and decay heat values are based on the earliest time of shipment. For the gas-cooled reactor technologies, the five-year time was selected because it is the current minimum allowed time before shipment per DOE contract. These values are compared with the reference LWR that used a 90-day decay time. Ninety days was the minimum allowed time before shipment for Table S-4. Since we are evaluating the transportation impacts, it is the inventory and associated decay heat at the time of shipment that is of interest, not the inventory and decay heat at any other particular time.

Comparisons should address [illegible] 1437 (5 yr min)

The fission product inventory at the time of shipment for the reference LWR was 6.19 x 10^6 Ci per MTU. The values for the fission product inventory at the time of shipment for the gas-cooled reactor technologies were both much lower, from 3.5 to 4 times lower.

— Basis ?

The actinide inventory at the time of shipment in Ci per MTU for the reference LWR was 1.42 x 10^5. Because of the longer burnup times for the new gas-cooled new reactor technologies, both of these reactor technologies have values that exceed the reference LWR. The GT-MHR and the PBMR, exceed the reference LWR by ~ 64% and ~59%, respectively. This comparison changes significantly for the GT-MHR if one considers the Ci per shipment, which is really what is of concern. The reference LWR ships 0.5 MTU per truck cask while the GT-MHR ships about a third less 0.16044 MTU per truck cask. Based on this comparison, the actinide inventory per shipment is about half (53%) for the GT-MHR versus the reference LWR. Since the PBMR plans to ship 0.495 MTU per cask, there is essentially no difference from the comparison per MTU.

really would [illegible] out Ci shipped per [illegible]

The total radioactive inventory in Ci per MTU at the time of shipment for the reference LWR was 6.33 x 10^6. The new gas-cooled reactor technologies have much lower total radioactivity at time of shipment. The differences are from 3 to almost 4 times lower.

5 yr vs. 90 day

The krypton-85 inventory in Ci per MTU at the time of shipment for the reference LWR was 1.13 x 10^4. Both the GT-MHR and the PBMR exceed the reference LWR by about a factor of 2.3. As before, if one considers the Ci per shipment, the Kr-85 inventory for the GT-MHR would be about 71% of the Kr-85 reference LWR inventory. The PBMR comparison remains essentially the same.

Ci/TRC's reactor

The kilowatts per MTU at the time of shipment for the reference LWR were 27.1. This value is considerably higher than for the gas-cooled reactor technologies. At the time of shipment, the decay heat for the gas-cooled reactor technologies being considered ranges from 6.36 kilowatts per MTU for the GT-MHR to 3.91 kilowatts per MTU for the PBMR.

5 yr vs. 2.9

The decay heat (per irradiated fuel truck cask in transit) in kilowatts for the reference LWR was 10. Both the gas-cooled reactor truck casks generate much less heat (5 to 10 times lower) per truck cask than the reference LWR.

The decay heat (per irradiated fuel rail cask in transit) in kilowatts for the reference LWR was 70. Since the gas-cooled reactor technologies are not planning to ship their spent fuel by rail, no comparison is needed. However, should DOE elect to transport by rail, the expected decay heat would be less than 70 based on the comparison for truck shipment.

At the time of the reference LWR evaluation, the road limit was 73,000 lbs. This has changed slightly through the years. 23 CFR 658.17 "Weight" states that for the Interstate and Defense Highways the maximum gross vehicle weight shall be 80,000 pounds. In all cases for the gas-cooled reactor technologies, the road limit is governed by state and federal regulations.

3.8.2.6 Discussion

NOT TRUE. Fuel/cladding could influence release. Enrichment determines source term.

Of the close to 30 characteristics/conditions that were examined, there are only 8 that were exceeded by the gas-cooled reactor technologies being considered. Three of these characteristics have no direct transportation impact on the health and the environment: fuel form, U_{235} enrichment, and fuel rod cladding. There are operational issues and fuel cycle impact issues associated with these characteristics that are addressed as part of the operating license and as part of the evaluation of Table S-3 "Uranium fuel cycle data,". respectively. Two of these characteristics (number of shipments for initial core loading and number of reload shipments) are really a part of the overall truck transportation picture. When one considers the total number of truck shipments (fresh fuel, spent fuel, and radioactive waste), the new reactor technologies have many fewer total shipments. For example, on an average annual basis, the new reactor technologies require 69 to 105 fewer truck shipments. Comparing the total number of shipments is appropriate since the radiological impacts from fresh fuel are negligible. One characteristic, burnup, manifests its impact through other characteristics, fuel inventory and decay heat at time of shipment, which are addressed separately. In the case of decay heat, both of the gas-cooled reactor technologies will generate fewer watts per MTU at time of shipment, and fewer kW per truck cask at time of shipment. The fuel inventory will be discussed as part of the remaining two characteristics that were exceeded: actinide inventory and krypton-85 inventory.

That the actinide inventory per metric ton of spent fuel is greater for the majority of the new gas-cooled reactor technologies is not surprising, since actinide activity tends to increase with increasing burnup and both of the gas-cooled reactor technologies plan a higher burnup than the reference LWR. The increase in the actinide activity for the new reactor technologies ranges from 59% to 65%. And as discussed in the previous section, if one considers the actinide inventory per shipment, only the PBMR exceeds the reference LWR by 59%. From NUREG/CR-6703 "Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU," we learn that "none of the actinides contributes more than one percent of the external dose from an iron transportation cask, and as a group, the actinides do not contribute significantly to the dose from transportation accidents. In fact, increasing the activities of Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-242 and Cm-244 by more than a factor of 1000 only increased the cumulative dose for a transportation accident during shipment of 43 GWd/MTU spent fuel from the northeast to Clark County, NV from 0.0358 to 0.0359 person-mSv/shipment (3.58×10^{-3} to 3.59×10^{-3} person-rem/shipment)." There is one other area where the increased actinide activity needs to be considered and that is the corresponding increase in neutron source term. NUREG/CR-6703 states "because neutrons are effectively attenuated by low-density materials such as plastics and water, it is believed that minor modifications can be made to shipping casks to allow them to transport the higher burnup fuel at full load."

Based on the analysis performed and the conclusions drawn in NUREG/CR-6703 which show that actinides are not major contributors to the transportation risk, either incident free or accident, and with the actinide activity only 59% greater, the environmental impacts would still be bounded even for these higher burnups.

This leaves the Kr-85 inventory as the final characteristic to be addressed. The increase of Kr-85, a long-lived noble gas, would suggest an increase of the consequences associated with an accident that resulted in a breach of the fuel cask and fuel rods. The range of increase for the gas-cooled technologies being considered is from 121% to 133%. And as discussed in the previous section, if one considers the Kr-85 inventory per shipment, only the PBMR exceeds the reference LWR. These amounts are based on a 5-year cooling time. If this decay time were increased by about 11 years, slightly greater than the half-life of Kr-85 (10.6 years), not an unlikely scenario by the way, this increase would for the most part decay away. Another factor to consider is that transportation risk is a function of both consequences and likelihood. Because the new reactor technologies require fewer truck shipments, the likelihood would decrease approximately 37% for the reactor with the greatest Kr-85 inventory. Another factor to consider is that the accident rate for large trucks has steadily declined for more than the past 25 years and is less than half the rate in 1975. Thus, the likelihood has decreased to about 37% (0.63 x 0.5) of the 1975 likelihood. A final and major factor to consider is that the cask regulations are based on allowable releases independent of the inventory. Thus, regardless of the initial source term, if the cask releases more than a specific acceptable amount, it would not be licensed. Based on these considerations, the 5-year Kr-85 quantities would still be bounded by the overall transportation risk profile provided by Table S-4.

3.8.2.7 Conclusion

In conclusion, this detailed comparison of the underpinnings of Table S-4 show that the existing environmental and health effects are still conservative and appropriate for use by the gas-cooled reactor technologies being considered. Of close to 30 characteristics examined, only eight were exceeded by the new technologies. In these instances, either they are independent of any impact or there are mitigating factors and controls to assure that these slight increases are bounded by the impacts specified in Table S-4. This conclusion is also borne out by the observation that these new reactor technologies will be using the same transportation modes and subject to the same NRC and DOT regulations for packaging and transportation as the original analysis that was used to develop Table S-4. Thus, the new reactor technologies under consideration and the transportation of radioactive material associated with them meet the conditions in 10 CFR 51.52(b).

3.8.3 Methodology Assessment

As indicated in Section 1.1.3, the selection of a reactor design to be used for the EGC ESP Facility is still under consideration. Selection of a reactor to be used at the EGC ESP Site may not be limited to those considered above. However, the methodology utilized above is appropriate to evaluate the final selected reactor. Further, should the selected design be shown to be bounded by the above evaluation, then the selected design would be considered to be within the acceptable transportation environmental impacts considered for this ESP.

D/N ADDRESS Releases From

References:

10CFR50.44, Standards for combustible gas control system in light-water-cooled power reactors
10CFR51.22, Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review
10CFR51.52, Table S-4 Environmental Impact of Transportation of Fuel and Waste
10 CFR 71, Packaging and Transportation of Radioactive Material
49 CFR 173, Shippers – General Requirements for Shipments and Packagings
49 CFR 178, Specifications for Packagings

Docket No. 50-400, 53 FR 30355 NRC Assessment of the Environmental Effects of Transportation Resulting From Extended Fuel Enrichment and Irradiation, August 11, 1988, and 53 FR 32322, August 24, 1988
NUREG-0170 Final Environmental Impact Statement on the Transportation of Radioactive Material by Air and Other Modes, Vols. 1 and 2, December 1977
NUREG-1437 Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Volumes 1 & 2, May 1996
NUREG-1555 Standard Review Plans for Environmental Reviews for Nuclear Power Plants, October 1999
NUREG/CR-6672 Reexamination of Spent Fuel Shipment Risk Estimates, March 2000
NUREG/CR-6703 Environmental Effects of Extending Fuel Burnup Above 60 Gwd/MTU, January 2001
WASH-1238 ENVIRONMENTAL SURVEY OF TRANSPORTATION OF RADIOACTIVE MATERIALS TO AND FROM NUCLEAR POWER PLANTS, December 1972
Supplement 1 to WASH-1238 (NUREG-75/038) ENVIRONMENTAL SURVEY OF TRANSPORTATION OF RADIOACTIVE MATERIALS TO AND FROM NUCLEAR POWER PLANTS, April 1975

Table 3.8-2 Gas-cooled Reactor Transportation Worksheet

Reactor Technology	Reference LWR (Single unit) (1100 MWe)	<u>GT-MHR</u> (4 Modules) (2400 MWt total) (1140 MWe total)	<u>PBMR</u> (8 Modules) (3200 MWt total) (1320 MWe total)	Comments
Characteristic				
Capacity	80%	88%	95%	
Normalization factor	1	0.88	0.7	
Reactor Power Level MWt	~ 3400	2400 (600 MWt per module, 4 modules per plant)	3200 (400 MWt per module, 8 modules per plant)	not exceeding 3800 MWt per reactor is a condition for use of Table S-4
Fuel Form	sintered UO_2 pellets	TRISO coated particle fuel with uranium oxycarbide (UCO) kernal	Sphere of TRISO Coated UO_2 fuel kernels	Sintered UO_2 pellets is a condition for use of Table S-4
U235 Enrichment	1% - 4%	fissile particle 19.8%; fertile particle natural uranium	initial 4.9%; equilibrium 12.9%	Not exceeding 4% is a condition for use of Table S-4; NUREG 1437 concludes that 5% is bounded
Fuel Rod Cladding	zircaloy	Graphite	Graphite	Zircaloy rods are a condition for use of Table S-4; 10 CFR 50.44 allows use of ZIRLO)

Gas-cooled Reactor Transportation Worksheet cont.

	Reference LWR (Single unit) (1100 MWe)	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total)	PBMR (8 Modules) (3200 MWt total) (1320 MWe total)	Comments
Characteristic				
Average burnup MWd/MTU	33,000	112,742	133,000	Not exceeding 33,000 is a condition for use of Table S-4; NUREG 1437 concludes 62,000 MWd/MTU for peak rod is bounded
Unirradiated fuel				
unirradiated fuel transport mode	truck	truck	truck	shipment by truck is a condition for use of Table S-4
# of shipments for initial core loading	18	51 shipments (1020 fuel elements per module x 4 modules; 80 elements per truck)	44 shipments (260,000 fuel spheres per module x 8 modules, 48,000 spheres per truck)	100 MTU for PWR; 150 MTU for BWR
# of reload shipments/year	6	20 shipments (520 elements per reload per 1.32 years x 4 modules; 80 elements per truck)	3 shipments (18,000 fuel spheres per module x 8 modules, 48,000 spheres per truck)	30 MTU annual reload

Gas-cooled Reactor Transportation Worksheet cont.

	Reference LWR (Single unit) (1100 MWe)	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total)	PBMR (8 Modules) (3200 MWt total) (1320 MWe total)	Comments
Irradiated fuel				
irradiated fuel transport mode	truck, rail or barge	truck	truck	shipment by truck, rail or barge is a condition for use of Table S-4
decay time prior to shipment	150 days	five years	five years	Not less than 90 days is a condition for use of Table S-4; 5 years is per contract with DOE
fission product inventory in Ci per MTU after 5 year decay	6.19×10^6	1.55×10^6	1.78×10^6	The value for the LWR is for a 90 day decay time.
Actinide inventory in Ci per MTU after 5 year decay	1.42×10^5	2.33×10^5	2.26×10^5	The value for the LWR is for a 90 day decay time.
Total radioactivity inventory in Ci per MTU after 5 year decay	6.33×10^6	1.78×10^6	2.01×10^6	The value for the LWR is for a 90 day decay time.
Krypton-85 inventory in Ci per MTU after 5 year decay	1.13×10^4	2.50×10^4	2.63×10^4	The value for the LWR is for a 90 day decay time.

Gas-cooled Reactor Transportation Worksheet cont.

	Reference LWR (Single unit) (1100 MWe)	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total)	PBMR (8 Modules) (3200 MWt total) (1320 MWe total)	Comments
Irradiated fuel				
watts per MTU after 5 year decay	2.71×10^4	6.36×10^3	3.91×10^3	The value for the LWR is for a 90 day decay time.
# of spent fuel shipments by truck	60	38 shipments (520 elements per module x 4 modules per 1.32 years, 42 elements per truck)	16 shipments (12 shipments for 1000 Mwe)	0.5 MT of irradiated fuel per cask
heat(per irradiated fuel truck cask in transit) kW	10	1.02 (6.356 kW/MTU x 0.16044 MTU/shipment)	1.9 (3.9 kw/MTU x .495 MTU/shipment)	
# of spent fuel shipments by rail	10	0	0	Appendix B, Table 1 says 3.2 MT of irradiated fuel per cask, Appendix B, Table 3 says 3.5
heat(per irradiated fuel rail cask in transit) kW	70	NA	NA	
# of spent fuel shipments by barge	5	0	0	

Gas-cooled Reactor Transportation Worksheet cont.

	Reference LWR (Single unit) (1100 MWe)	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total)	PBMR (8 Modules) (3200 MWt total) (1320 MWe total)	Comments
radioactive waste				
radioactive waste transport mode	truck or rail	truck	truck	Shipment by truck or rail is a condition for use of Table S-4
# of rad waste shipments by truck	46	6 (1100 Ci/yr; 98 m^3/yr)	9 (800 drums)	assumed 90% of the waste shipped at 1000 ft^3 per truck, 10% at 200 ft^3 per truck
weight per truck lbs.	73,000	governed by state and federal regulations	governed by state and federal regulations	current interstate gross vehicle limit is 80,000 lbs.(23 CFR 658.17)
# of rad waste shipments by rail	11	0	0	
weight per cask per rail car tons	100	100	100	
Transport totals				
traffic density, trucks per day	less than 1	less than 1	less than 1	
rail density, cars per month	less than 3	0	0	

Gas-cooled Reactor Transportation Worksheet cont.

Yellow indicates a value
larger than or different from
the reference LWR

Notes:
The results for the reactor technologies have not been adjusted for their larger electrical generation or increased capacity factor.

References:
10CFR51.52, Table S-4 Environmental Impact of Transportation of Fuel and Waste

Table 3.8-3

Summary Table S-4-Environmental Impact of Transportation of Fuel and Waste To and From One Light-Water-Cooled Nuclear Power Reactor[1]

Normal Conditions of Transport

Condition
Value

Heat (per irradiated fuel cask in transit)
250,000 Btu/hr.

Weight (governed by Federal or State restrictions)
73,000 lbs. Per truck; 100 tons per cask per rail car.

Traffic density:

Truck
Less than 1 per day.

Rail
Less than 3 per month.

Exposed Population
Estimated Number of Persons Exposed
Range of Doses to Exposed Individuals[2] (per reactor year)
Cumulative Dose to Exposed Population (per reactor year)[3]

Transportation workers — 200

0.01 to 300 millirem
4 man-rem.

General public:

Onlookers — 1,100

0.003 to 1.3 millirem
3 man-rem.

Along Route — 600,000

0.0001 to 0.06 millirem

Accidents in Transport

Types of Effects
Environmental Risk

Radiological effects

Small[4]

Common (nonradiological) causes
1 fatal injury in 100 reactor years; 1 nonfatal injury in 10 reactor years; $475 property damage per reactor year.

[1]Data supporting this table are given in the Commission's "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," WASH-1238, December 1972 , and Supp. 1 NUREG-75/038 April 1975. Both documents are available for inspection and copying at the Commission's Public Document Room, 2120 L Street NW., Washington, DC and may be obtained from the National Technical Information Service, Springfield, VA 22161 . WASH-1238 is available form NTIS at a cost of $5.45 (microfiche, $2.25) and NUREG-75-038 is available at a cost of $3.25 (microfiche $2.25).

[2]The Federal Radiation Council has recommended that the radiation doses from all sources of radiation other than natural background and medical exposures should be limited to 5,000 millirem per year for individuals as a result of occupational exposure and should be limited to 500 millirem per year for individuals in the general population. The dose to individuals due to average natural background radiation is about 130 millirem per year.

[3]Man-rem is an expression for the summation of whole body doses to individuals in a group. Thus, if each member of a population group of 1,000 people were to receive a dose of 0.001 rem (1 millirem), or if 2 people were to receive a dose of 0.5 rem (500 millirem) each, the total man-rem dose in each case would be 1 man-rem.

[4]Athough the environmental risk of radiological effects stemming from transportation accidents is currently incapable of being numerically quantified, the risk remains small regardless of whether it is being applied to a single reactor or a multireactor site.

ER SECTION 7.4
TRANSPORTATION ACCIDENTS

7.4 Transportation Accidents

The assessment of transportation accidents is provided in Section 3.8, Transportation of Radioactive Materials.

ER SECTION 5.7
URANIUM FUEL CYCLE IMPACTS

5.7 Uranium Fuel Cycle Impacts

This section addresses the uranium fuel cycle environmental impacts and is divided into two main subsections. The first subsection addresses the light-water-cooled reactor (LWR) designs presently being considered. The second subsection addresses the gas-cooled reactor designs also being considered. This split addresses the regulatory distinction made in 10 CFR 51.51 for light-water-cooled reactors.

5.7.1 Light-water-cooled Reactors

10 CFR 51.51(a) states that "Every environmental report prepared for the construction permit stage of a light-water-cooled nuclear power reactor, and submitted on or after September 4, 1979 shall take Table S-3, *Table of Uranium Fuel Cycle Environmental Data*, as the basis for evaluating the contribution of the environmental effects of uranium mining and milling, the production of uranium hexafluoride, isotopic enrichment, fuel fabrication, reprocessing of irradiated fuel, transportation of radioactive materials and management of low level waste and high level wastes related to uranium fuel cycle activities to the environmental costs of licensing the nuclear power plant. Table S-3 shall be included in the environmental report and may be supplemented by a discussion of the environmental significance of the data set forth in the table as weighed in the analysis for the proposed facility."

Table S-3 of 10 CFR 51.51 is reproduced in its entirety herein as Table 5.7-3. Specific categories of natural-resource use included in the table relate to land use, water consumption and thermal effluents, radioactive releases, burial of transuranic and high- and low-level wastes, and radiation doses from transportation and occupational exposures. The contributions in the table for reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle); that is, the cycle that results in the greater impact is used.

Descriptions of the environmental impact assessment of the uranium fuel cycle as related to the operation of light-water-cooled reactors are well documented by the USNRC. The environmental impact of a light-water-cooled reactor on the U.S. population from radioactive gaseous and liquid releases (including radon and technetium) due to the uranium fuel cycle is small when compared with the impact of natural background radiation. In addition, the nonradiological impacts of the uranium fuel cycle are acceptable.

The light-water-cooled reactor technologies being considered are identified in Section 1.1.3. These LWR designs include the ABWR (Advanced Boiling Water Reactor), the ESBWR (Economic Simplified Boiling Water Reactor), the AP-1000 (Advanced Passive PWR), the IRIS (International Reactor Innovative and Secure), and the ACR-700 (Advanced light-water-cooled version of the CANDU Reactor). The standard configuration for each of these reactor technologies is as follows. The ABWR is a single unit, 4300 MWt, 1500 MWe reactor. The ESBWR is a similar BWR: single unit, 4000 MWt, 1390 MWe. The AP-1000 is a single unit, 3400 MWt, 1117-1150 MWe pressurized water reactor. The IRIS is a three module pressurized water reactor configuration for a total of 3000 MWt and 1005 MWe. And the ACR-700 is a twin unit, 3964 MWt, 1462 MWe, light-water-cooled CANDU reactor.

These reactor technologies are all light-water-cooled nuclear power reactors with uranium dioxide fuel and therefore Table S-3 of paragraph (a) of 10 CFR 51.51 with the current amendment (as given in 49 FR 9381, March 12, 1984 and 49 FR 10922, March 23, 1984) provides the environmental effects from the uranium fuel cycle for these reactor technologies.

5.7.2 Gas-cooled Reactors

5.7.2.1 Introduction and Background

This section provides an assessment of the environmental impacts of the fuel cycle, as related to the operation of the gas-cooled reactor technologies, based on a comparison of the key parameters that were used to generate the impacts listed in 10 CFR 51.51 Table S-3 (and repeated in Table 5.7-3). The key parameters are energy usage, material involved, number of shipments, etc. associated with the major fuel cycle activities. These activities are mining and milling, uranium hexafluoride conversion, enrichment, fuel fabrication, and radioactive waste disposal. Basically, the premise is that if less energy is needed, if fewer shipments are required, and if less material is involved in the process, then with all other things being equal, the overall impacts are less.

There are two gas-cooled reactor technologies being considered at this time. The GT-MHR is a four module, 2400 MWt, 1140 MWe reactor that operates at a unit capacity of 88%. The PBMR is an eight module, 3200 MWt, 1320 MWe reactor operating at a 95% unit capacity.

A key reference is NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, May 1996, which provides a very detailed look at the impacts to the environment from the nuclear fuel cycle. The document also looks at the sensitivity of the changes to the nuclear fuel cycle on the impacts to the environment. As these changes are much more representative of the current and future situation than what was considered in the WASH-1248 *Environmental Survey of the Uranium Fuel Cycle* report, the conclusions of NUREG-1437 will be used in the following discussion.

Table 5.7-1, "The Gas-Cooled Fuel Cycle Worksheet" was prepared to succinctly capture the major features of the reference LWR fuel cycle that were used to develop Table S-3 and compare these same features with the gas-cooled reactor technologies being considered. This comparison can then help to demonstrate that the existing Table S-3 is appropriate for use by these technologies. The premise being that if the values of the major contributors to the health and environmental impacts that were used for the reference LWR fuel cycle are greater than those comparable values for the gas-cooled reactor technologies then the published impacts would also be greater and suitable for use by the new reactor technologies. It is important to point out that even though we are looking at the contributors individually, it is the overall impact that is of concern. As such, there can be increases in individual contributors, yet the total impacts can still be bounded, if offset by decreases in other contributors.

The information to construct the worksheet was taken from 10 CFR 51.51 Table S-3 "Uranium Fuel Cycle Environmental Data," WASH-1248 *Environmental Survey of the Uranium Fuel Cycle*, and Supplement 1 to WASH-1248 (also known as NUREG-0116) *Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle*. The "reference LWR" refers to the model 1000 MWe light-water-cooled nuclear reactor used as a basis for studying annual fuel related requirements as described in WASH-1248. For the gas-cooled reactor technologies, information was gathered from the reactor vendors, United States Enrichment Corporation (USEC) and ConverDyn.

5.7.2.2 Analytic Approach

The major activities of the reference LWR fuel cycle that were considered in the WASH-1248 report were uranium mining, uranium milling, uranium hexafluoride production, uranium enrichment, fuel fabrication, irradiated fuel reprocessing, radioactive waste management which includes decontamination and decommissioning, and transportation. Three comments pertinent to this analysis are: 1) the WASH-1248 report and this evaluation only address the uranium fuel cycle (other fuel cycles such as thorium and plutonium are not part of this effort), 2) irradiated fuel reprocessing is not being considered by any of the new reactor technologies and is not included in this analysis, and 3) the transportation impacts are addressed based on the following premise - if the quantity of material required by the new gas-cooled reactor technologies at each major step of the fuel cycle is less than the reference plant, then the transportation impacts are also less. Comparing only the number of shipments of material is appropriate since there is little if any radioactivity in the fuel cycle shipments considered by Table S-3.

The main features of the major activities of the reference LWR fuel cycle that were identified as being the primary contributors to the health and environmental impacts are as follows. For the mining operation, annual ore supply is the major determinant of environmental and health impacts. Less ore will necessitate less energy, fewer emissions, less water usage, and less land disturbed. Secondarily, the mining technique can play a significant role in any impacts. Open pit mining has by far the most environment impact, followed by underground mining, with *in situ* leaching being the most environmentally benign.

For the milling operation, annual yellowcake (U_3O_8) production is the metric of interest. If a plant requires less U_3O_8 than the reference plant, then there will be less energy needed, fewer emissions, and less water usage. This is especially true if *in situ* leaching was used to obtain the ore, because the major milling steps of crushing and grinding are not required.

For the uranium conversion process, annual uranium hexafluoride (UF_6) production is the primary determinant of environmental impacts. If the new technology requires less UF_6

than the reference plant, then there will be less energy required, fewer emissions and less water used. As with the mining step, the conversion process (wet versus dry) is also a consideration. However, NUREG 1437 states that in either case "the environmental releases are so small that changing from 100 percent use of one process to 100 percent of the other would make no significant difference in the totals given in Tables S-3 or S-4."

For the enrichment operation, there are two quantities of interest. The first quantity is the separative work units (SWU) needed to enrich the fuel, and the second quantity is the amount of enriched UF_6. The SWU is a measure of energy required to enrich the fuel. More SWUs would by itself indicate not only more energy required but also more emissions associated with the production of the energy needed and with that more water usage. However, this assumes the same technology is used to achieve the enrichment. As discussed in NUREG 1437, the centrifuge process uses 90 percent less energy than the gaseous diffusion process. Since the major environmental impacts for the entire fuel cycle are from the emissions from the fossil fueled plants needed to supply the energy demands of the gaseous diffusion plant, this reduction in energy requirements results in a fuel cycle with much less environmental impact. With regard to the amount of enriched UF_6 produced, the major effect would be the number of shipments. More UF_6 would necessitate more shipments, while less UF_6 would require fewer shipments. Slight increases or decreases would probably result in the same number of shipments.

For the fuel fabrication process, the quantity of UO_2 produced is the value of interest. This is really equivalent to the annual fuel loading in MTU, which will also be evaluated. Here again, the production of more UO_2 would require more energy, greater emissions, and increased water usage. New reactor technologies with an annual fuel loading less than the reference LWR plant would have less environmental impact, requiring less energy, fewer emissions and less water usage.

The last activity to be addressed is radioactive waste management. There are two aspects of radioactive waste that are considered as part of Table S-3: operations and reactor decontamination and decommissioning (D&D). For these activities, curies (Ci) of low-level waste (LLW) from annual operations and Ci of LLW from reactor (D&D) are the measures to consider. Curies by themselves are not a direct indicator of the potential environmental impacts. The radionuclide, its half-life and type of emission, and its physical and chemical form are the main contributors to risk. While we recognize this distinction, for this bounding analysis we will use curies as was done in the WASH-1248. More curies generally indicate the potential for greater impacts, while fewer curies indicate lesser impacts.

One of the clearest ways to conduct this comparison between the reference LWR and the gas-cooled reactor technologies is to start with the annual fuel loading in MTU for each of the reactor technologies. The other activities more accurately originate from the need for a certain amount of fuel. Using annual fuel loading as the starting point, the analysis will proceed in the reverse direction for the fuel cycle until the mining has been addressed, then the radioactive waste will be addressed. Before beginning this comparison, it is important to recognize that the plants being considered are a different size, have a different electrical rating and have a different capacity factor from the reference LWR. The reference LWR is a 1000 MWe plant with a capacity factor of 80%. In order to make a proper comparison, we need to evaluate the activities based on the same criterion. In this case, electrical generation is the metric of choice. Electrical generation is why the plants are being built and we want to know if these new reactor technologies, for the same electrical output, have a greater or lesser impact on human health and environment. Based on this, the reactor technologies will be normalized to 800 MWe using plant specific electrical rating and capacity factor.

5.7.2.3 Analysis and Discussion

5.7.2.3.1 Fuel Fabrication / Operations

The reference LWR required 35 MTU on an annual basis. This is equivalent to 40 MT of enriched UO_2, the annual output needed from the fuel fabrication plant. In comparison, the normalized annual fuel needs for the new gas-cooled reactor technologies ranged from 4.3 MTU to 5.3 MTU, approximately 88% to 85% lower than the reference plant. Similarly, the annual output needed from the fuel fabrication plant range from a low of 4.89 MT of UO_2 to 6.0 MT of UO_2, again approximately 88% to 85% lower than the reference plant. The specific breakdowns are shown on Table 5.7-1. One important distinction is that the fuel form for the gas-cooled reactors is also different. For the GT-MHR, the fuel is a two-phase mixture of enriched UO_2 AND UC_2, usually referred to as UCO. For the PBMR the fuel kernel is UO_2. Both fuels are then TRISO coated. For the GT-MHR these TRISO fuel particles are blended and bonded together with a carbonaceous binder. These fuel compacts are then stacked within a graphite block. For the PMBR, the fuel unit is a 6 cm diameter graphite sphere containing approximately 15000 fuel particles.

Before concluding the potential impacts from the fuel fabrication process are less, the gas-cooled reactors require a different fuel fabrication process altogether. The TRISO coated fuel kernel is quite different from the UO_2 sintered fuel pellet and as such would require a different type of facility. Ideally, to ensure the environmental impacts of this change in fabrication process are bounded by the reference LWR fuel fabrication plant, a comparison of the land use, energy demand, effluents, etc., is in order. However, because there are no planned or currently operating plants in the United States, a direct comparison cannot be made at this time. Therefore, we have provided information on the reference fuel fabrication plant along with conceptual design information for a TRISO fabrication plant that was planned for the New Production Reactor and conceptual design information received from one of the gas-cooled reactor vendors.

From WASH-1248, the reference LWR fuel fabrication plant produced fuel for 26 plants (~910 MTU), was located on a site of about 100 acres, required 5.2 million gallons of water per annual fuel requirement of 35 MTU, and required 1,700 MW-hours of electricity per 35 MTU. The WASH-1248 report also states that nearly all of the airborne chemical effluents resulted from the combustion of fossil fuels to produce electricity to operate the fabrication plant. These numbers represented a very small portion of the overall fuel cycle. For example, the electrical usage represented less than 0.5% of that needed for the enrichment process, and the water use was less than 2% of the overall fuel cycle.

The fuel fabrication facility for the New Production Reactor was for a modular high temperature gas reactor (MHTGR) design and was sized for just one plant, so any comparisons with the much larger reference LWR fuel fabrication plant are problematic. The dimensions for the fuel fabrication building were 230 ft x 150 ft. The annual production was about 2 MTU. The plant required 960 kW of electrical power and 45 liters per minute of water. Effluents consisted of 60 m^3/yr of miscellaneous non-combustible solids and filters; 50 m^3/yr of combustible solids; 50 m^3/yr of process off-gas and HVAC filters; 2.0 m^3/yr of tools and failed equipment; and process off-gases of 900,000 m^3/yr. The process off-gases consisted of 74 % N_2, 12% O_2, 7.2% Ar, 6.4% CO_2, 0.2% CO, and 0.02% CH_3CCl_3. The activity associated with this off-gas: 0.01 pCi alpha/m^3, and 0.01 pCi beta/m^3.

The information gathered from one of the current reactor vendors was for a plant producing 6.3 MTU, about 19% more than the annual reload of 5.31 MTU for its reactor. Again this plant was sized for just one reactor. This plant would require 10 MW of electrical power with an annual electrical usage of 35,000 MW-hr. The gaseous emissions consist of 80 MT of nitrogen, 52 MT of argon, 22.4 MT of CO, 22 MT of hydrogen and 3.7 MT of CO_2. The solid waste totals about 84 m^3 of LLW, 3 m^3 of intermediate level waste, and the remainder sanitary/industrial wastes. The liquid processing system would generate an additional 3.8 m^3 of LLW, would discharge about 3700 m^3 of low activity aqueous effluent, and would discharge about 45,000 m^3 of industrial cooling water.

Because of the differences in scale and the state of design of the facilities, it is not possible or appropriate to make a direct comparison of the impacts. Obviously, there are economies of scale and design improvements that will occur for a plant comparable in size to the reference plant. Regardless, the projected impacts of a TRISO fuel plant based on the two conceptual designs are not inconsistent with the reference plant and would be operated within existing air, water, and solid waste regulations. Further; like the impacts associated with the sintered UO_2 pellet plant, the impacts from a TRISO fuel plant would still be a minor contributor to the overall fuel cycle impacts. By characterizing the impacts as "not inconsistent," we mean that while certain parameters such as electrical usage for fuel fabrication might be higher for the gas-cooled plants on an annual fuel loading basis, the environmental impacts from the TRISO plants as conceptualized would still be bounded by the overall LWR fuel cycle impacts.

5.7.2.3.2 Uranium Enrichment

In order to produce the 40 MT of enriched UO_2 for the reference LWR, the enrichment plant needed to produce 52 MT of UF_6, which required 127 MT of SWU. The normalized enriched UF_6 needs for the new gas-cooled reactor technologies ranged from 6.38 MT of UF_6 to 7.9 MT of UF_6, approximately 88% to 85% lower. To produce these quantities of UF_6 requires from 124 MT of SWU to 163 MT of SWU, slightly lower to 28% higher. The enrichment SWU calculation for the new reactor technologies was performed using the USEC SWU calculator and assumes a 0.30% tails assay, the same value as for the NUREG-0116 reference plant. Using this calculator for the reference LWR plant yielded 126 MT of SWU versus the NUREG value of 127. This is very close indicating that this latest version of the USEC SWU calculator is appropriate for use in this computation. Table 5.7-2 "Gas-cooled Reactor SWU and Feed Calculation Results" gives the details of the computations.

The 28% increase in the MTU of SWU would by itself indicate greater environmental impacts. However, a close look at the original WASH 1248 analysis shows that the environmental impacts are almost totally from the electrical generation needed for the gaseous diffusion process. These impacts result from the emissions from the electrical generation that is assumed to be from coal plants and from the associated water to cool the plants. Today, and in the future, the enrichment process is and will be different. A significant fraction of the enrichment services to US utilities today is provided from European facilities using centrifuge technology rather than the fifty-year-old gaseous diffusion technology. For the future, two private companies, United States Enrichment Corporation and Louisiana Energy Services, are planning to develop centrifuge technology in the US. In fact, NRC has just recently accepted United States Enrichment Corporation's centrifuge license application for technical review. Centrifuge technology requires less than 10% of the energy needed for the gaseous diffusion process and as such the environmental impacts associated with the electrical generation will be correspondingly less. This tremendous reduction in energy and the associated environmental impacts more than offsets a 28% increase in SWU.

5.7.2.3.3 Uranium Hexafluoride Production

In order to provide the feed needed for the reference LWR to the enrichment plant, the uranium hexafluoride plant needed to produce 360 MT of UF_6. The normalized feed needed for the new gas-cooled reactor technologies, the output from the uranium hexafluoride plant, ranged from 241 to 303 MT of UF_6, well below the reference plant. The feed calculations were performed using the USEC SWU calculator. Using this calculator for the reference LWR yielded 353 MT of UF_6 versus the NUREG value of 360. Again this value is very close (<2%) to the published value.

5.7.2.3.4 Uranium Milling

To produce the 360 MT of UF_6 for the reference LWR, 293 MT of yellowcake (U_3O_8) from the mill was required. The normalized new gas-cooled reactor technologies needs ranged from 193 MT of U_3O_8 to 243 U_3O_8, well below the reference plant. These yellowcake numbers were generated using the relationship 2.61285 lbs of U_3O_8 to 1 kg of UF_6. This conversion factor was obtained from ConverDyn.

5.7.2.3.5 Uranium Mining

The raw ore needed to produce the 293 MT of yellowcake (U_3O_8) for the reference LWR was 272,000 MT. Now assuming a 0.1% ore body and a 90% recovery efficiency, the normalized new gas-cooled reactor technologies ore requirements ranged from 215,000 to 270,000 MT of ore, both below the reference plant. Of note, the NUREG table value of 272,000 should be about 325,600 using the same assumptions. It is not clear why this number is different, but in any case, the gas-cooled reactor technologies are below the published reference plant value.

Uranium mining completes the front end of the fuel cycle. However, there are two areas on the down stream cycle to be considered. These are the LLW generated by operations and the LLW generated as part of the D&D process. As mentioned earlier, spent fuel reprocessing is not germane to this analysis, and therefore, not discussed.

5.7.2.3.6 Solid Low-Level Radioactive Waste - Operations

For the reference LWR, 10 CFR 51.51, Table S-3, Table of Uranium Fuel Cycle Environmental Data, states that there are 9,100 Ci of LLW generated annually from operations. The range of activity of LLW generated annually projected by the new gas-cooled reactor technologies is 65.4 Ci to 1,100 Ci, far below the reference LLW. This decrease would also suggest many fewer shipments to the disposal facility and less worker exposure.

5.7.2.3.7 Solid Low-Level Radioactive Waste – Decontamination and Decommissioning

10 CFR 51.51, Table S-3, states 1,500 Ci per Reactor Reference Year (RRY) "comes from reactor decontamination and decommissioning – buried at land burial facilities." Based on this small quantity and the modifying phrase "buried at land burial facilities" it is clear that only waste suitable for shallow land burial is being considered. At this time, only general conclusions can be drawn to indicate these gas-cooled reactor technologies would generate less D&D LLW than the reference plant. The new plants will operate much cleaner than the reference LWR as evidenced by the annual generation of much less LLW. Improvements in fuel integrity and differences in fuel form as well as the use of the chemically and radiologically inert helium as the coolant are responsible for this reduction and also should contribute to both a lower level and less overall contamination to be managed during the D&D process. The plants higher thermal efficiency and higher fuel burnup would produce less heavy metal radioactive waste. Lastly, the plants are typically more compact than the reference LWR contributing to less D&D waste. Of note, the entry for the PBMR indicated approximately 15 times the RRY curie quantity of D&D waste. The main radionuclides identified for this waste are Co-60 and Fe-55 with half-lives of 5.26 years and 2.73 years respectively. Based on these half-lives, after about 20 years the activity would be less than the reference LWR.

5.7.2.4 Summary and Conclusion

To recap, there are only two instances where any part of the uranium fuel cycle is/might be exceeded by the new gas-cooled reactor technologies. These fuel cycle steps are enrichment, a 28% increase and possibly D&D. As discussed above, the enrichment requirement for SWU, while slightly larger, can be conducted in a much more environmentally benign manner, centrifuge versus gaseous diffusion, from current overseas sources or expected new domestic facilities. The net effect will be that the environmental and health impacts will be less than those identified in Table S-3. The second area, decontamination and decommissioning, is a minor contributor to the fuel cycle impacts. A slight increase in the D&D step is more than offset by the significant decreases in the impacts due to reduction in fuel needs and changes in the enrichment process and mining technique.

In conclusion, this detailed comparison of the underpinnings of Table S-3 show qualitatively that the existing WASH-1248 environmental and health effects are still conservative and appropriate for use by these new gas-cooled reactor technologies. Collectively, improvements in both past practices as well as changes in technology have resulted in a fuel cycle with lower environmental impact.

5.7.3 Methodology Assessment

As indicated in Section 1.1.3, the selection of a reactor design to be used for the ESP Facility is still under consideration. Selection of a reactor to be used at the ESP Site may not be limited to those considered above. However, the methodology utilized above is appropriate to evaluate the final selected reactor. Further, should the selected design be shown to be bounded by the above evaluation, then the selected design would be considered to be within the acceptable fuel cycle environmental impacts considered for this ESP.

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Table 5.7-1 Gas-cooled Fuel Cycle Worksheet

Reactor Technology Facility/Activity	Reference LWR (Single unit) (1000 MWe) 80% Capacity	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total) 88% Capacity	PBMR (8 Modules) (3200 MWt total) (1320 MWe total) 95% Capacity
Mining Operations			
Annual ore supply MT	272,000	337140	337140
Normalized annual ore supply MT	272,000	269712	214739
fraction of reference LWR	1	0.99	0.79
Calculated number	314,011	269712	214739
Milling Operations			
Annual yellowcake MT	293	303	303
Normalized annual yellowcake MT	293	243	193
fraction of reference LWR	1	0.83	0.66
Calculated number	283	243	193
UF_6 Production			
Annual UF_6 MT	360	379	379
Normalized annual UF_6 MT	360	303	241
fraction of reference LWR	1	0.84	0.67
Calculated number	353	303	241

Table 5.7-1 Gas-cooled Fuel Cycle Worksheet cont.

Facility/Activity — Reactor Technology	Reference LWR (Single unit) (1000 MWe) 80% Capacity	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total) 88% Capacity	PBMR (8 Modules) (3200 MWt total) (1320 MWe total) 95% Capacity
Enrichment Operations			
Enriched UF_6 MT	52	8.0	12.3
Normalized enriched UF_6 MT	52	6.38	7.9
fraction of reference LWR	1	0.12	0.15
Calculated number	52	6.38	7.9
Annual SWU MT	127	204	194
Normalized annual SWU MT	127	163	124
fraction of reference LWR	1	1.29	0.97
Calculated number	126	163	124
Fuel Fabrication Plant Operations			
Enriched UO_2 MT	40	6.11	9.5
Normalized enriched UO_2 MT	40	4.89	6.0
fraction of reference LWR	1	0.12	0.15
Calculated number	40	4.89	6.0
Annual Fuel Loading MTU	35	5.39	8.34
Normalized annual fuel loading MTU	35	4.3	5.31
fraction of reference LWR	1	0.12	0.15

Table 5.7-1 Gas-cooled Fuel Cycle Worksheet cont.

Facility/Activity	Reactor Technology: Reference LWR (Single unit) (1000 MWe) 80% Capacity	GT-MHR (4 Modules) (2400 MWt total) (1140 MWe total) 88% Capacity	PBMR (8 Modules) (3200 MWt total) (1320 MWe total) 95% Capacity
Reprocessing Plant Operations			
Annual spent fuel reprocessing MTU	35	0	0
Solid Radioactive Waste			
Annual LLW from reactor operations Ci	9,100	1100 Ci; 98 m^3	65.4 Ci; 800 drums
fraction of reference LWR	1	0.12	0.01
LLW from Reactor Decontamination & Decommissioning Ci per RRY	1,500	[redacted]	2.2×10^4 (5.30×10^5 Ci after 24 years operation and 2 years decay)
TRU and HLW Ci	1.1×10^7	NA	NA

Table 5.7-1 Gas-cooled Fuel Cycle Worksheet cont.

Yellow indicates a value
larger than Table S-3

Blue indicates data is
missing or incomplete

References:
10CFR51.51, Table S-3 Table of Uranium Fuel Cycle Environmental Data

Notes:
The enrichment SWU calculation was performed using the USEC SWU calculator and assumes a 0.30% tails assay.
The information on the reference reactor (mining, milling, UF6, enrichment, fuel fabrication values) taken from NUREG-0116, Table 3.2, no recycling
The information on the reference reactor (solid radioactive waste) taken from 10CFR51.51, Table S-3
The calculated information on the reference reactor uses the same methodology as for the reactor technologies.
The normalized information is based on 1000 MWe and the reactor vendor supplied unit capacity factor.
For the new reactor technologies, the annual fuel loading was provided by the reactor vendor.
The USEC SWU calculator also calculated the kgs of U feed. This number was multiplied by 1.48 to get the necessary amount of UF_6.
The annual yellowcake number was generated using the relationship 2.61285 lbs of U_3O_8 to 1 kg U of UF_6; 1.185 kgs of U3O8 to 1.48 kg of UF6
The annual ore supply was generated assuming an 0.1% ore body and a 90% recovery efficiency.
Co-60 with a 5.26 year half-life and Fe-55 with a 2.73 year half-life are the main nuclides listed for the PBMR D&D waste.

Table 5.7-2 Gas-cooled Reactor SWU and Feed Calculation Results

Reactor Technology	Kgs Uranium Product	Weight Percent U235	SWU Quantity	Kgs of U Feed Required	Tails Assay
GT-MHR	5,394	19.80%	204373	255918	0.30%
PBMR	8,340	12.90%	194414	255679	0.30%
NUREG 0116	35,000	3.10%	126,175	238,455	0.30%
WASH 1248	35,000	3.20%	147,280	223,965	0.25%

Notes:
The reactor vendor supplied the Kgs uranium product and weight percent U235.
The tails assay was assumed to be 0.3% to match NUREG-0116 with the exception of WASH 1248 which used a tail assay of 0.25%
The SWU Quantity and Kgs Feed Required were calculated using the USEC SWU Calculator
The results have not been normalized to equivalent electrical generation.

Table 5.7-3, 10 CFR 51.51 Table S-3- of Uranium Fuel Cycle Environmental Data[1]

[Normalized to model LWR annual fuel requirement [WASH-1248] or reference reactor year [NUREG-0116]]

[See Footnotes at end of this table]

Environmental Considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
Natural Resource Use		
Land (acres)		
Temporarily committed[2]	100	
Undisturbed area	79	
Disturbed area	22	Equivalent to a 110 MWe coal-fired power plant.
Permanently committed	13	
Overburden moved (millions of MT)	2.8	Equivalent to 95 MWe coal-fired power plant.
Water (millions of gallons)		
Discharged to air	160	=2 percent of model 1,000 MWe LWR with cooling tower.
Discharged to water bodies	11,090	
Discharged to ground	127	
Total	11,377	<4 percent of model 1,000 MWe LWR with once through cooling.
Fossil Fuel:		
Electrical energy (thousands of MW-hour)	323	<5 percent of model 1,000 MWe output
Equivalent coal (thousands of MT)	118	Equivalent to the consumption of a 45 MWe coal-fired power plant.
Natural gas (millions of scf)	135	<0.4 percent of model 1,000 MWe energy output.
Effluents-Chemical (MT)		
Gases (including entrainment):[3]		
SO_x	4,400	
NO_x[4]	1,190	Equivalent to emissions from 45 MWe coal-fired plant for a year.
Hydrocarbons	14	
CO	29.6	
Particulates	1,154	
Other gases		
F	.67	Principally from UF_6, production, enrichment, and reprocessing. Concentration within range of state standards- below level that has effects on human health.
HCl	.014	
Liquids:		
SO_4^-	9.9	From enrichment, fuel fabrication, and reprocessing

NO_3^-	25.8	steps. Components that constitute a potential for adverse environmental effect are present in dilute concentrations and receive additional dilution by receiving bodies of water to levels below permissible standards. The constituents that require dilution and the flow of dilution water are: NH_3-600cfs., NO_3-. 20cfs., Fluoride-70cfs.
Fluoride	12.9	
CA^{++}	5.4	
$C1^-$	8.5	
Na^+	12.1	
NH_3	10.0	
Fe	.4	
Tailings Solutions (thousands of MT)	240	From mills only-- no significant effluents to environment.
Solids	91,000	Principally from mills-- no significant effluents to environment.
Effluents-- Radiological (curies)		
Gases (including entrainment):		
Rn-222		Presently under reconsideration by the Commission.
Ra-226	.02	
Th-230	.02	
Uranium	.034	
Tritium (thousands)	18.1	
C-14	24	
Kr-85(thousands)	400	
Ru-106	.14	Principally from fuel reprocessing plants.
I-129	1.3	
I-131	.83	
Tc-99		Presently under consideration by the Commission
Fission products and transuranics	.203	
Liquids:		
Uranium and daughters	2.1	Principally from milling-- included tailings liquor and returned to ground -- no effluents; therefore, no effect on the environment.
Ra-226	.0034	From UF_6 production.
Th-230	.0015	
Th-234	.01	From fuel fabrication plants-- concentration 10 percent of 10 CFR 20 for total processing 26 annual fuel requirements for model LWR.
Fission and activation products	5.9×10^{-6}	
Solids (buried on site):		
Other than high level (shallow)	11,300	9,100 Ci comes from low level reactor wastes and 1,5000 Ci comes from reactor decontamination and decommissioning -- buried at land burial facilities. 600 Ci comes from mills -- included in tailing returned to ground. Approximately 60 Ci comes from conversion and spent fuel storage. No significant effluent to the environment.

TRU and HLW (deep)	1.1×10^7	Buried at Federal Repository
Effluents-- thermal (billions of British thermal units)	4,063	<5 percent of model 1,000 MWe LWR.
Transportation (person-rem):		
Exposure of workers and general public	2.5	
Occupational exposure	22.6	From reprocessing and waste management.

[49 FR 9381, Mar. 12, 1984; 49 FR 10922, Mar. 23, 1984]

[1] In some cases where no entry appears it is clear from the background documents that the matter was addressed and that, in effect, the Table, should be read as if a specific zero entry had been made. However there are other areas that are not addressed at all in the Table. Table S-3 does not include health effects from the effluents described in the Table, or estimates of releases of Radon-222 from the uranium fuel cycle or estimates of Technetium-99 released from waste management or reprocessing activities. These issues may be the subject of litigation in the individual licensing proceedings.

Data supporting this table are given in the Environmental Survey of the Uranium Fuel Cycle," WASH-1248, April 1974; the "Environmental Survey of Reprocessing and Waste Management Portion of the LWR Fuel Cycle," NUREG-0116 (Supp. 1 to WASH-1248); the "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," NUREG-0216 (Supp.2 to WASH-1248): and in the record of final rulemaking pertaining to Uranium Fuel Cycle Impacts from Spent Fuel Reprocessing and Radioactive Waste Management, Docket RM-50-3. The contributions from reprocessing, waste management and transportation of wastes are maximized for either of the two fuel cycles (uranium only and fuel recycle). The contribution from transportation excludes transportation of cold fuel to a reactor and of irradiated fuel and radioactive wastes from a reactor which are considered in Table S-4 of §51.20(g). The contributions from the other steps of the fuel cycle are given in columns A-E of Table S-3A of WASH-1248.

[2] The contributions to temporarily committed land from reprocessing are not prorated over 30 years, since the complete temporary impact accrues regardless of whether the plant services one reactor for one year or 57 reactors for 30 years.

[3] Estimated effluents based upon combustion of equivalent coal for power generation.

[4] 1.2 percent from natural gas use and process.

ACRONYMS

The following list gives the major acronyms and abbreviations that were used in the ER sections and supporting documentation.

Acronyms

ABWR	Advanced Boiling Water Reactor
ACR-700	Advanced CANDU Reactor
AECL	Atomic Energy of Canada, Limited
AP-1000	Advanced Passive Pressurized Water Reactor
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
CFR	Code of Federal Regulations
D&D	Decontamination and Decommissioning
DOE	U. S. Department of Energy
DOE-NE	DOE Office of Nuclear Energy, Science and Technology
DOT	U. S. Department of Transportation
ESBWR	Economic Simplified Boiling Water Reactor
ESP	Early Site Permit
FR	Federal Register
GT-MHR	Gas Turbine-Modular Helium Reactor
INEEL	Idaho National Engineering and Environmental Laboratory
IRIS	International Reactor Innovative and Secure
ISL	*in situ* leaching
kW	kilowatt
LEU	Low Enriched Uranium
LLW	Low-level Radioactive Waste
LWR	Light Water Reactor
MT	Metric Ton
MTU	Metric Ton Uranium
MWd	Megawatt days
MWe	Megawatt electric
MWt	Megawatt thermal
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Regulation
PBMR	Pebble Bed Modular Reactor
PWR	Pressurized Water Reactor
RRY	Reactor Reference Year
SECY	NRC Office of the Secretary
SWU	Separative Work Unit
TRISO	Fuel kernal coating – three layers of pyrolytic carbon, one layer of silicon carbide
UCO	uranium oxycarbide
USEC	United States Enrichment Corporation

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Chapter 6 Environmental Impacts of Transportation describes potential impacts of transportation activities nationally and in Nevada on the transportation-related affected environment
Appendix J Transportation provides the basis for potential impacts related to national and Nevada transportation, as discussed in Chapter 6.
Appendix M Supplemental Transportation Information in response to public comments, this appendix provides general information not specifically related to the transportation analysis considered in Chapter 6 and Appendix J.

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PRESENTATIONS

This section provides copies of the three slide presentations that were given to the NRC on September 25, 2002; December 5, 2002; and January 29, 2003.

Background

- Tables S-3 & S-4 used to assess environmental impacts for model LWR
- S-3 - impacts from uranium fuel cycle
- S-4 - impacts from transportation of new and spent fuel and waste
- Use of tables required for use in preparing an Early Site Permit
- Basis documents for S-3 and S-4 are WASH 1248/NUREG 0116 and WASH 1238/NUREG-75-038

Characteristics of the advanced reactor designs that are consistent with Tables S-3 and S-4

- Use of NRC and DOT licensed casks
- Acceptable risk levels
- Modes of transportation

Characteristics of the advanced reactor designs that affect the comparison with Tables S-3 and S-4

- Reprocessing
- Enrichment
- Burn-up
- Fuel types
- Cooling time prior to shipment
- Demographics
- Current accident statistics
- Waste disposal

Proposed Approach

- Understand the critical assumptions, parameter values and basis used to develop the current values in Tables S-3 and S-4
- Update assumptions and data sources
- Gather comparable parameter values from the reactor types under consideration
- Develop environmental impacts from advanced reactor types

Proposed Approach (cont.)

- For those impacts bounded by the existing table, document the results
- For those impacts that are not bounded, revisit the assumptions and data used, and prepare alternative values for use by advanced reactor types
- Above all, the environmental impacts should be equivalently protective

Schedule

- Support the existing Early Site Permit plans
- Interim Status meeting with NRC in December
- Draft evaluation by the end of January
- Final report by end of April
- Repond to NRC questions as needed

Summary

- Proposed approach for verifying the use of Tables S-3 & S-4 for advanced reactor designs being considered under the DOE Near Term Deployment initiative

ESP-8

Methodology for Estimating Fuel Cycle and Transportation Environmental Impacts for Early Site Permit Applications

December 5, 2002

NEI

ESP-8 Objectives

- Update NRC staff on industry's Tables S-3/ S-4 initiative
- Original update intended to provide preliminary results
- Due to revised (earlier) meeting date, this briefing provides additional details regarding methodology
- This briefing also describes approach if certain assumptions in existing tables do not initially bound new technologies

NEI

Proposed Methodology for Determining Fuel Cycle Environmental Impacts

- Determine fuel cycle requirements [uranium, enrichment, transportation] for range of technologies considered by ESP applications
- Compare fuel cycle requirements to those used to develop Tables S3 and S4
- Where the fuel cycle requirements are lower than the conditions assumed to develop Tables S3 and S4, use the current table impacts for the environmental evaluation
- Where any fuel cycle requirements are higher than the conditions assumed to develop Tables S3 and S4, evaluate potential impacts along with other fuel cycle technology changes that may have reduced environmental impacts

NEI

10 CFR 51.51, Table S-3

- Table S-3 developed based on fuel requirements for a model 1000 MWe LWR
- Uranium, SWU, and transport requirements will be compared with the values used as basis of current Table S-3 for the same energy output
- Technology improvements that have tended to reduce environmental impacts may offset any increase in fuel cycle and transportation requirements

NEI

Fuel Cycle Technology Changes

- Higher fuel burnup
  - Reduces average annual fuel loading [lower number of fuel assemblies at higher enrichment]
  - Generally reduces average annual uranium ore requirements, but may slightly increase SWU
- Higher Operating Plant Capacity Factor
  - Increases both energy production and fuel requirements
- Improved enrichment processes
  - Lower emissions from electric generation
  - Improved energy efficiency [especially for centrifuge enrichment technology]
- No spent fuel reprocessing expected

NEI

10 CFR 51.52, Table S-4

- Current Table S-4 is based on the transportation of fuel and waste to and from a 1100 MWe LWR subject to the following conditions
  - Core power not to exceed 3,800 MWt
  - Uranium dioxide pellets of less than 4% enrichment encapsulated in zircaloy rods
  - Average irradiation of no more than 33,000 megawatt-days per metric ton, and no assembly shipped until at least 90 days after discharge

NEI

10 CFR 51.52, Table S-4 (cont.)

- The number, modes, types and radioactive inventories of shipments of spent fuel and wastes will be determined for a range of reactor technologies and compared to the values used as a basis of current Table S-4 for the same energy output
- Any increases of these values will be evaluated
- Technology improvements have tended to reduce transportation environmental impacts and may off set any changes in transportation conditions

NEI

Changes in Fuel Cycle Transportation Technology

- Higher fuel burnup reduces spent fuel generation and reduces quantity of spent fuel to be shipped
- New fuel types do not all use zircaloy rods
- Longer cooling time after discharge [minimum of 5 years - average of over 10 years] reduces source term at transport
- Transport casks for new fuel types and higher burnup fuel must meet same normal and accident dose limits

NEI

ESP-8, Tables S-3 & S-4

Fuel Cycle and Transportation Evaluation of New Reactor Technologies

January 29, 2003

NEI

ESP-8 Task History

- September 25th - presented an approach to Tables S-3 & S-4
- Gathered background information, vendor data, other supporting materials
- December 5th - presented refined methodology
- January 29th – discussion of preliminary findings

NEI

Key Points of the Methodology

- As in the WASH reports, use conservative but reasonable assumptions
- Compare ESP fuel cycle and transportation requirements with those assumed to calculate the environmental impacts shown in Tables S-3 & S-4
- Evaluate any potential increases in fuel cycle requirements [e.g., enrichment]
- Demonstrate that Tables S3 & S4 are suitable for determining the expected fuel cycle environmental impacts in ESP applications

NEI

General Observations

- Stricter environmental regulations are in effect for all operations
- Mining and milling operations are considerably different
- UF_6 conversion similar
- Enrichment process potentially much different

NEI

Matter before the ESP Task Force

- Are the existing Tables S-3 and S-4 bounding and appropriate for advanced reactor types?
- If not, what would be equivalent environmental effects
- Six reactor types currently being considered: 3 advanced LWRs, 2 gas-cooled reactors and 1 advanced heavy water

Early Site Permit Approach to Tables S-3 & S-4

Nuclear Energy Institute

Early Site Permit Task Force

Presentation to the

U. S. Nuclear Regulatory Commission

September 25, 2002

V-104

General Observations (cont.)

- Fuel fabrication similar for light water reactors, different for gas-cooled reactors
- Low-level waste generation from operations much less
- Transportation regulations similar
- Recent evaluations still support conclusion that transportation impacts are minimal

NEI

Preliminary Fuel Cycle Results

- Generally the new reactor technologies require less uranium ore, yellowcake, and UF_6 so the mining, milling, and conversion impacts should be bounded
- Slightly higher SWU in some cases, up to 20% in one case, but due to changes in enrichment technology, stricter environmental regulations and method of electrical generation the fuel cycle environmental impacts shown in Table S3 are still appropriate for the ESP applications

NEI

Preliminary Fuel Cycle Results (cont.)

- Annual fuel loading exceeded in one case but the planned number of shipments is 2 fewer than the reference LWR so the impacts are expected to be bounded
- Much less LLW from operations so radwaste impacts would be bounded
- Still evaluating D&D and gas-cooled fuel fabrication

NEI

Preliminary Transportation Results

- Thermal power exceeded in one case; potential impacts addressed as part of the overall fuel cycle
- Fuel form, cladding different in two cases; potential impacts addressed as part of the packaging requirements
- Enrichment and burnup exceeded in one case; potential impacts addressed as part of the packaging requirements

NEI

Preliminary Transportation Results (cont.)

- Initial core loading shipments exceeded in two cases; potential impacts bounded since the total number of shipments (initial and annual reload) are less than the reference LWR
- Fuel inventory is greater in some cases; potential impacts addressed as part of the packaging requirements

NEI

ESP-8 Summary

- Preliminary results indicate that fuel cycle and transportation impacts for a range of new reactor technologies are consistent with Tables S3 and S4
- Preparing to send ESP-8 resolution letter
- NRC staff feedback desired on industry-proposed approach

NEI

VENDOR DATA SUBMITTALS

The following section is divided into seven subsections for the seven reactor technologies that were considered. The seven sections are the ABWR, ESBWR, AP-1000, IRIS, ACR-700, GT-MHR and the PBMR. Along with the vendor data submittals are subsequent e-mail discussions that modified some of the original supplied data.

ABWR

"Beard, James A. (PS, NE)" <james.beard@gene.GE.com>

02/13/2003 09:46 AM

To: "Beard, James A. (PS, NE)" <james.beard@gene.GE.com>, rln@inel.gov
cc: "Cambria, Michael (Parsons)" <Michael.Cambria@parsons.com>, "Smith, Marvin (Dominion)" <Marvin_Smith@dom.com>, "Semmes, Spencer (Dominion)" <Spencer_Semmes@dom.com>, "Atambir S. Rao (PS, NE) (E-mail)" <atambir.rao@gene.GE.com>
Fax to:
Subject: RE: ABWR Decay Heat Loads for 5 year Old Fuel

All:

Let me try again with the information in an attached file and not as an embeeded object.

Alan

<<5 year decay heat.doc>>

> -----Original Message-----
> From: Beard, James A. (PS, NE)
> Sent: Thursday, February 13, 2003 8:27 AM
> To: 'rln@inel.gov'
> Cc: 'Cambria, Michael (Parsons)'; 'Smith, Marvin (Dominion)'; 'Semmes,
> Spencer (Dominion)'
> Subject: ABWR Decay Heat Loads for 5 year Old Fuel
>
> Robert:
>
> Please revise the information regarding decay heat of 5 year old fuel that
> we provided previously with the information below.
>
> Let me know if you have any furhter questions.
>
> g GE NUCLEAR ENERGY
> ______________________________
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>
> Program Manager
> James.Beard@gene.ge.com
> Work Phone (301) 208-1460 or (408) 925-3524
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>
>
>
> << OLE Object: Device Independent Bitmap >>

5 year decay heat.d

Table 5-2, Decay Heat after Five Years Cooling (no uncertainty allowance)

	Initial Core Discharge	Reload Core Discharge
Relative Decay Heat after 5 yrs	7.800×10^{-5}	9.886×10^{-5}
Decay Heat (MW/MTU)	2.14×10^{-3}	2.71×10^{-3}

Table 5-3, Decay Heat after Five Years Cooling (two sigma uncertainty allowance)

	Initial Core Discharge	Reload Core Discharge
Relative Decay Heat after 5 yrs	8.398×10^{-5}	1.060×10^{-4}
Decay Heat (MW/MTU)	2.30×10^{-3}	2.90×10^{-3}

"Beard, James A. (PS, NE)" <james.beard@gene.GE.com> on 02/12/2003 08:56:50 AM

To: rln@inel.gov
cc:

Subject: FW: Spent Fuel Radioactivity

> Bob:
>
> Attached is a table with the radioactive inventory of the ABWR spent fuel
> 5 years after discharge from the reactor. Let me know if you have any
> questions or concerns.
>
> Alan
>
>
>
> <<CoreInventory.doc>>
>
> Provide estimates of the spent fuel inventories and radioactivity, in Ci
> per MTU, after 5 years of decay
> • Fission product inventory
> • Actinide inventory
> • Total radioactivity
> • Krypton-85 inventory
>
> The reactor type is an ABWR at a power level of 4300 MWt
> The fuel type is GE 14 with the following characteristics.

	Initial Core	Reload 1	Reload 2 to Eq.
Core Size, number of bundles	872	872	872
Core Thermal Power, MWth	4300	4300	4300
Operating Cycle Length, days	605	605	605
Operating Capacity Factor, %	100	100	100
Refueling Outage Duration, days	30	30	30
Refueling Interval, months	21	21	21
Loaded Batch Size	872	240	316
Batch Average Enrichment, w/o U235	3.5	4.5	4.5
Average Bundle Mass, KgU	180	180	180
Batch Average Burnup, GWd/MT	36	46	46

CoreInventory.d

Five Year Decay Inventory, GE Fuel

Activation Products		Actinides + Daughters		Fission Products	
Isotope	Curie/MTU	Isotope	Curie/MTU	Isotope	Curie/MTU
Ag-109m	7.76E-04	Am-241	1.34E+03	Ag-108	3.44E-06
Ar-37	5.48E-16	Am-242	3.32E+01	Ag-109m	1.36E-04
Ar-39	3.42E-04	Am-242m	3.34E+01	Ag-110	4.44E-01
C-14	7.70E-01	Am-243	3.24E+01	Ag-110m	3.34E+01
Ca-41	1.18E-03	Am-245	2.50E-09	Ba-137m	1.18E+05
Ca-45	8.65E-04	Bi-212	4.49E-02	Cd-113m	6.13E+01
Cd-109	7.76E-04	Bk-249	1.72E-04	Cd-115m	6.79E-10
Cd-115m	1.12E-13	Cm-241	1.01E-17	Ce-141	1.49E-11
Cl-36	1.86E-02	Cm-242	5.51E+01	Ce-144	1.14E+04
Co-58	7.49E-05	Cm-243	3.69E+01	Cs-134	4.81E+04
Co-60	2.73E+03	Cm-244	4.86E+03	Cs-135	8.22E-01
Cr-51	5.28E-16	Cm-245	6.56E-01	Cs-137	1.24E+05
Eu-152	1.08E-03	Cm-246	1.41E-01	Eu-152	1.09E+01
Eu-154	1.53E+02	Np-235	5.01E-04	Eu-154	1.01E+04
Eu-155	7.14E+01	Np-237	6.16E-01	Eu-155	5.22E+03
Fe-55	3.35E+03	Np-238	1.67E-01	Gd-153	1.41E-01
Fe-59	4.32E-10	Np-239	3.24E+01	H-3	5.34E+02
Gd-153	2.26E+01	Np-240m	9.23E-07	I-129	4.20E-02
H-3	5.24E-04	Pa-233	6.16E-01	In-114	4.21E-11
Hf-175	3.05E-07	Pa-234m	3.13E-01	In-114m	4.39E-11
Hf-181	1.08E-10	Pb-212	4.49E-02	In-115m	4.77E-14
Ho-166m	2.39E-02	Po-212	2.88E-02	Kr-85	8.90E+03
In-113m	1.95E-02	Po-216	4.49E-02	Nb-93m	7.54E-01
In-114	1.63E-09	Pu-236	3.56E-01	Nb-95	6.78E-03
In-114m	1.70E-09	Pu-237	4.37E-12	Nb-95m	2.27E-05
Ir-192	8.59E-08	Pu-238	6.14E+03	Pd-107	1.46E-01
K-42	1.63E-12	Pu-239	3.87E+02	Pm-146	1.84E+00
Lu-177	9.04E-07	Pu-240	6.15E+02	Pm-147	3.37E+04
Lu-177m	3.93E-06	Pu-241	1.22E+05	Pm-148	8.59E-11
Mn-54	3.46E+01	Pu-242	2.24E+00	Pm-148m	1.52E-09
Mo-93	1.95E-02	Pu-243	5.85E-07	Pr-144	1.14E+04
Nb-93m	1.98E-01	Ra-224	4.49E-02	Pr-144m	1.37E+02
Nb-94	1.76E-01	Rn-220	4.49E-02	Rh-102	4.67E-01
Nb-95	4.55E-04	Th-228	4.49E-02	Rh-103m	1.09E-08
Nb-95m	1.52E-06	Th-231	2.20E-02	Rh-106	1.64E+04
Ni-59	2.59E+00	Th-234	3.13E-01	Ru-103	1.21E-08
Ni-63	4.20E+02	Tl-208	1.61E-02	Ru-106	1.64E+04
P-32	2.89E-08	U-232	6.00E-02	Sb-124	1.17E-06
Re-188	9.56E-08	U-234	1.47E+00	Sb-125	4.45E+03
Ru-103	8.38E-16	U-235	2.20E-02	Sb-126	1.43E-01
S-35	1.84E-05	U-236	3.77E-01	Sb-126m	1.02E+00
Sb-124	2.93E-08	U-237	3.00E+00	Se-79	5.61E-01
Sb-125	9.16E+02	U-238	3.13E-01	Sm-151	5.60E+02

Five Year Decay Inventory, GE Fuel

Activation Products

Isotope	Curie/MTU
Sc-46	7.04E-07
Sn-113	1.95E-02
Sn-119m	5.20E+01
Sn-121m	1.12E+00
Sn-123	2.40E-02
Sr-89	8.94E-10
Sr-90	6.68E-03
Ta-182	1.74E-01
Tb-160	1.49E-03
Tc-99	4.80E-03
Te-123m	2.45E-04
Te-125m	2.24E+02
Te-127	7.52E-07
Te-127m	7.67E-07
Tm-170	2.43E-07
W-181	1.82E-04
W-185	1.08E-05
W-188	9.46E-08
Y-90	6.68E-03
Y-91	6.98E-08
Zn-65	2.39E-03
Zr-93	6.86E-01
Zr-95	2.05E-04
Total	7.98E+03

Actinides + Daughters

Isotope	Curie/MTU
U-240	9.23E-07
Total	1.36E+05

Fission Products

Isotope	Curie/MTU
Sn-119m	1.24E+00
Sn-121m	2.46E-01
Sn-123	1.60E-01
Sn-126	1.02E+00
Sr-89	8.33E-06
Sr-90	8.85E+04
Tb-160	3.14E-05
Tc-99	1.74E+01
Te-123m	6.79E-04
Te-125m	1.09E+03
Te-127	1.02E-01
Te-127m	1.04E-01
Te-129	1.02E-12
Te-129m	1.57E-12
Y-90	8.85E+04
Y-91	3.38E-04
Zr-93	2.50E+00
Zr-95	3.05E-03
Total	5.87E+05

129

TOTAL - 171

Five Year Decay Inventory, GE Fuel

Activation Products

Isotope	Curie/MTU
Ag	7.78E-04
Ar	3.42E-04
Be	2.12E-06
C	7.70E-01
Ca	2.05E-03
Cd	7.76E-04
Cl	1.86E-02
Co	2.73E+03
Cr	5.28E-16
Eu	2.24E+02
Fe	3.35E+03
Gd	2.26E+01
H	5.24E-04
Hf	1.01E-06
Ho	2.39E-02
I	1.75E-13
In	1.95E-02
Ir	1.70E-07
K	3.20E-08
Lu	4.84E-06
Mn	3.46E+01
Mo	1.95E-02
Nb	3.75E-01
Ni	4.23E+02
Os	7.29E-10
P	2.89E-08
Pb	8.97E-08
Re	2.45E-07
Ru	2.66E-14
S	1.84E-05
Sb	9.16E+02
Sc	7.04E-07
Si	2.88E-08
Sm	4.19E-06
Sn	5.32E+01
Sr	6.68E-03
Ta	1.74E-01
Tb	1.49E-03
Tc	4.80E-03
Te	2.24E+02
Tm	7.28E-06
V	2.01E-14

Actinides + Daughters

Isotope	Curie/MTU
Am	1.44E+03
Bi	4.49E-02
Bk	1.72E-04
Cm	4.95E+03
Np	3.32E+01
Pa	9.30E-01
Pb	4.49E-02
Po	7.36E-02
Pu	1.29E+05
Ra	4.49E-02
Rn	4.49E-02
Th	3.80E-01
Tl	1.61E-02
U	5.24E+00
Total	1.36E+05

Fission Products

Isotope	Curie/MTU
Ag	3.38E+01
Ba	1.18E+05
Cd	6.13E+01
Ce	1.14E+04
Cs	1.72E+05
Eu	1.53E+04
Gd	1.41E-01
H	5.34E+02
Ho	4.51E-03
I	4.20E-02
In	9.98E-11
Kr	8.90E+03
La	1.47E-10
Nb	7.61E-01
Nd	2.17E-09
Pd	1.46E-01
Pm	3.37E+04
Pr	1.15E+04
Rb	2.99E-05
Rh	1.64E+04
Ru	1.64E+04
Sb	4.45E+03
Se	5.61E-01
Sm	5.60E+02
Sn	2.66E+00
Sr	8.85E+04
Tb	3.14E-05
Tc	1.74E+01
Te	1.09E+03
Xe	6.58E-17
Y	8.85E+04
Zr	2.50E+00
Total	5.87E+05

Five Year Decay Inventory, GE Fuel

Activation Products

Isotope	Curie/MTU
W	1.93E-04
Y	6.68E-03
Zn	2.39E-03
Zr	6.86E-01
Total	7.98E+03

Actinides + Daughters

Isotope	Curie/MTU

Fission Products

Isotope	Curie/MTU

"Beard, James A. (PS, NE)" <james.beard@gene.GE.com>

02/04/2003 06:21 PM

To: "Cambria, Michael (Parsons)" <Michael.Cambria@parsons.com>, eddie.grant@exeloncorp.com, rln@inel.gov
cc:
Fax to:
Subject: ABWR S-3 and 4 Information

Mike, Eddie and Robert:

Here at long last is the S-3 and S-4 information. I hope that we have provided the information that you need to complete your assessment. If you should require any additional information please let me know right away.

Thanks

Alan

g GE NUCLEAR ENERGY

J. Alan Beard

Program Manager
James.Beard@gene.ge.com
Work Phone (301) 208-1460 or (408) 925-3524
Cell Phone (301) 461-3497

<<S3_S4 Questions-ABWR.doc>>

S3_S4 Questions-ABWR

ESP 8: Reactor Vendor Questionnaire

Information on Annual Fuel Requirements

1. Define Standard Technical Configuration.
   - Provide expected reactor power, MW_t and MW_e for each reactor

     For the GE ABWR and ESBWR the uprated thermal power of the ABWR of 4300 MWt is used to bound both reactors. For reference the currently certified power level of the ABWR is 3926 MWt and the ESBWR design value is 4000 MWt

   - Number of modules or reactors expected for a typical unit configuration for small modular systems

     The ABWR and ESBWR are both designed as single unit plants.

2. Expected Fuel Loading
   - Provide Initial Core Fuel Loading in MTU

     The initial core load for the ABWR and ESBWR are approximately equal. For the ABWR the initial core load is 156.96 MT of Uranium.

   - Provide Annual Average Fuel Loading in MTU based on 40 years of operation

     The average annual fuel loading in MTU is 32.76. This is based on an average capacity factor of 95%. This capacity factor is subject to variation by the operating practices of the utility but in GE's view represents a reasonably achievable measure. This figure includes allowances for refueling and maintenance outages but does not include any provision for extended outages.

   [Note: Provide the basis for the above estimates, i.e. estimated unit capacity factor, refueling/maintenance outage frequencies and durations, and average expected energy produced per year.]

3. Average Fuel Enrichment in % U-235
   [Note: Provide table of MTU and enrichment if multiple fuel enrichments are normally used for the initial core or fuel reloads]

   The batch average enrichment of the core is less than 3.5% for the initial core and less than 4.5% for the subsequent reloads.

4. Fuel form
   - Provide Fuel Assembly (or Basic Fuel Unit) Drawing

     See Figure at back of information

- Provide a Table giving the following for each fuel unit:

Total Mass	Bundle average mass is 266 kg (without channel) Bundle average mass is 298 kg (with channel)
Uranium Mass	Bundle Average Uranium mass 180 kg
Volume	
Outside Dimensions	14.2 cm x 14.2 cm x 447.0 cm (with channel)

An estimate of the typical number of fuel assemblies or units required for the initial core and the average expected number of fuel assemblies or units per year for core reloads

For the ABWR the core holds 872 fuel assemblies. The information provided in response to this request is based on the GE-14 fuel type, which is the latest offering of the GE fuels group. The basic design of the GE-14 is the same as earlier BWR fuels offered by GE as far as overall dimensions. However, improvements in the design have been made to optimize the fuel utilization.

The ESBWR core will hold a total of 1020 bundles. The cross sectional area of these bundles will be the same as the ABWR. However they are approximately 15% shorter so the net effect is that the same amount of Uranium is held in the core as for the ABWR.

5. Fuel materials
   - Provide a table of fuel material types and mass for a typical fuel unit including a description of fuel, structural, and cladding materials

The channel, fuel rods (cladding), water rods, spacers and end plugs are all fabricated from Zircalloy, of which there is approximately 85 kg used in each bundle.

The upper and lower tie plates and assorted fasteners are fabricated from stainless steel of which there is approximately 6.8 kg used in each bundle

There are a number of small components that are fabricated from inconel of which there is a approximately 0.5 kg used per bundle.

87
4.4 / 131
872 / 1003

Information on Annual Fuel Requirements (cont'd)

6. Define the expected typical transport mode (i.e. truck, rail, etc.) for delivery of the unirradiated fuel from the fabrication facility or port of entry to the reactor site

   Typical shipment of new fuel from the GE fuel facility in Wilmington, NC is by flat bed tractor trailer.

7. Provide a general description of the transport containers expected to be used for delivery of unirradiated fuel
   - Capacity of each container, i.e. number of fuel units per container

     The transport containers consist of a dual packaging system. Two fuel assemblies are first packed in a padded steel box. The steel box is then packaged inside a padded wooden crate. The dimensions of a typical wooden crate are 30" x 30" x 15'6"

   - Number of containers that can be transported on one legal weight truck shipment

     The number of bundles that typically can be shipped on a single truck is either 28 or 30 and is limited by weight.

[Note: This data is intended to allow for a determination of the number of shipments and MTU for the initial core loading and the average number of shipments and MTU per year for core reloads.]

Information on Expected Low Level Waste Production

1. Estimated annual average LLW production expected from reactor operations
   - Provide an estimate of the expected volumes and curies of LLW

     The production of LLW is in large part controlled by the practices of the owner. GE in the design certification chose not to establish unreasonable expectations for future owners and as such followed the maximum target values. In this case the volume of LLW is 100 cubic meters per year with an estimated curie content of 2700 Ci.

2. LLW expected from reactor decontamination and decommissioning
   - Provide an estimate of the expected volumes and curies of LLW produced due to reactor decontamination and decommissioning

The process for decontamination and decommissioning of an ABWR is outside the control of GE and subject to a great deal of variation depending on the timing and the methods chosen. As such, GE is unable to provide a reasonable estimate for these values.

Information on Spent Fuel Production/Transport

1. Spent Fuel Shipments
   - Provide an estimate of the quantity (MTU) of irradiated fuel that can be transported in one legal weight truck cask [25 ton cask] or typical rail cask [100 ton cask], assuming 5 year cooling after discharge.
     [Note: Estimate should be in MTU (based on unirradiated MTU) and number of fuel units. to allow for a determination of the average number of spent fuel shipments expected per year of reactor operation.]

     GE is not familiar with the constraints of fuel assemblies that can be transported in the commercially available casks. What we can tell you is the GE-14 fuel type is nearly identical to the other GE fuel types and the number of BWR fuel assemblies that can be shipped should not be different for the ABWR.

     Assume ? BWRs

2. Provide the average fuel burnup in MWd/MTU

   After achieving an equilibrium core, the batch average burnup is 46 GWd/MT

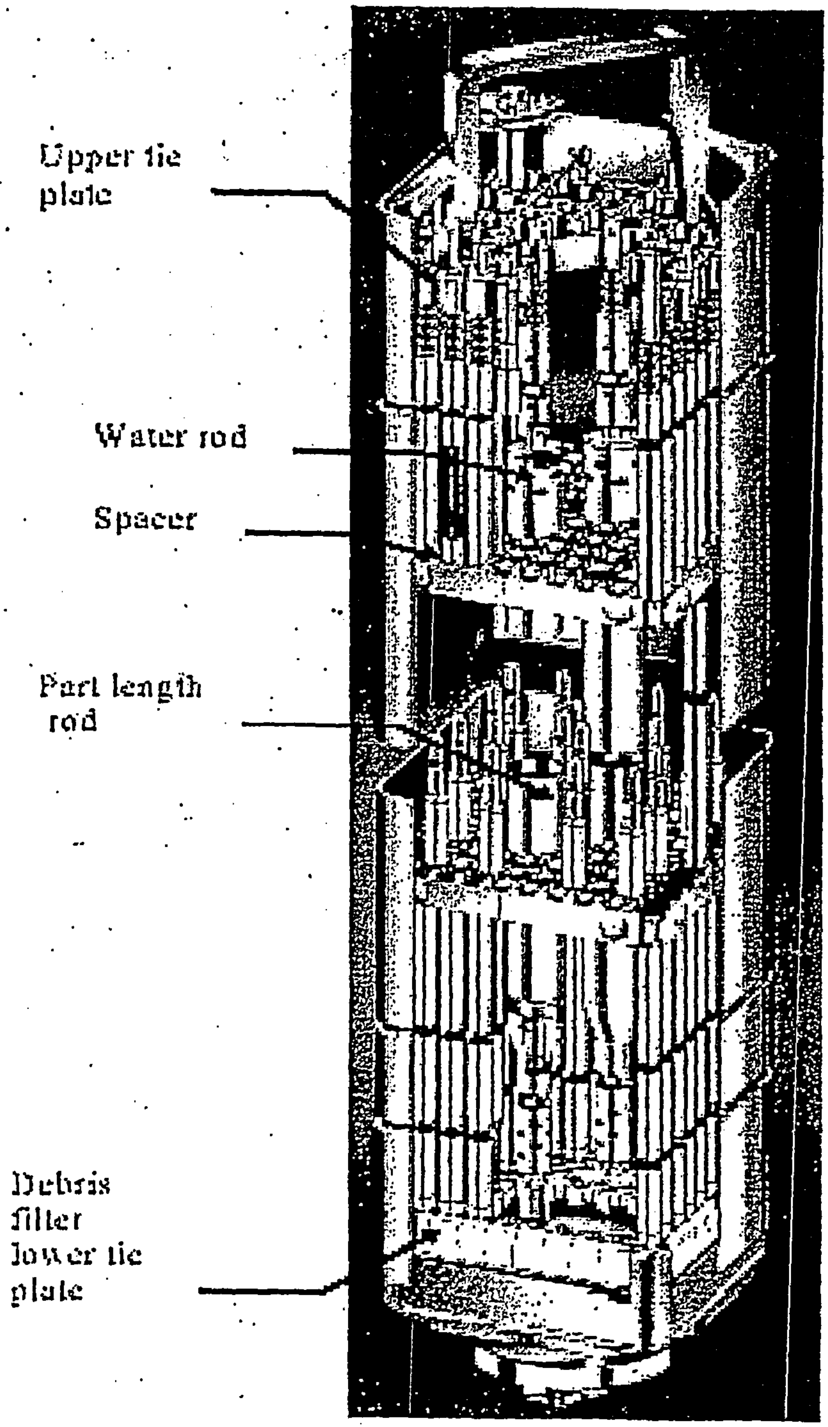
3. Provide an estimate of the decay heat in watts per MTU after 5 years of decay from fuel discharge

   The estimated decay heat per MTU 5 years after discharge from the core is between 18-22 kilowatts.

4. Provide estimates of the spent fuel inventories and radioactivity, in Ci per MTU, after 5 years of decay
   - Fission product inventory
   - Actinide inventory
   - Total radioactivity
   - Krypton-85 inventory

   GE is still trying to gather these numbers and will provide this information as soon as possible.

   [Note: If available, please provide a complete set of the ORIGEN run results (or other applicable code for the appropriate reactor type) detailing the spent fuel inventories at 5 years decay to answer questions 3 and 4.]


Upper tie plate
Water rod
Spacer
Part length rod
Debris filter lower tie plate

ESBWR

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com> on 03/17/2003 01:29:04 PM

To: RLN@inel.gov
cc:

Subject: RE: ESP-8 Information for the ESBWR

We originally developed the design for the European market but now are in the pre-certification phase with the NRC for US certification. The "E" officially stands for "Economic". And yes, the "S" is simplified.

Sorry for the long winded answer.
Roy

-----Original Message-----
From: RLN@inel.gov [mailto:RLN@inel.gov]
Sent: Monday, March 17, 2003 12:31 PM
To: Challberg, Roy C. (PS, NE)
Subject: RE: ESP-8 Information for the ESBWR

Thanks for the confirmation Roy. One more little item. What do the letters ES of the initials ESBWR represent? I have seen European Simplified and Economic Simplified but most of the time it escapes definition. Thanks for your help.

Bob

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com>
03/17/2003 12:39 PM

To: RLN@inel.gov
cc:
Fax to:
Subject: RE: ESP-8 Information for the ESBWR

Bob-
You're exactly right. The bounding decay heat value for one of the reload cores for ABWR (4300 MWt) was 2.9 kW. That was the decay heat after 5 years
(with a 2 sigma uncertainty). This will bound the 4000 MWt ESBWR fuel.
Roy

-----Original Message-----
From: RLN@inel.gov [mailto:RLN@inel.gov]
Sent: Monday, March 17, 2003 9:09 AM
To: Challberg, Roy C. (PS, NE)
Subject: RE: ESP-8 Information for the ESBWR

Thanks much Roy. I understand the potential variability of the numbers. For this effort, we just need some justification for the value we select, so thanks again.

In looking closer at the data, one other item has arisen: the decay heat value. I know for the ABWR they originally gave the same 18 -22 kW per MTU value. Later, upon questioning , it was modified to 2.9. This was the two sigma uncertainty value for the reload core discharge after five years cooling.

If you would please check into this for the ESBWR.

Thank you.

Bob

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com>
03/17/2003 09:43 AM

To: RLN@inel.gov
cc:
Fax to:
Subject: RE: ESP-8 Information for the ESBWR

Bob-
Our recent heat balance of our total plant design for a typical site gives us 1390 MWe, which of course is highly dependent upon site conditions and type of heat sink.
Let me know if you need anything else.
Roy

GE Nuclear Energy
Advanced Reactor Projects
(408) 925-3317

-----Original Message-----
From: RLN@inel.gov [mailto:RLN@inel.gov]
Sent: Friday, March 14, 2003 3:19 PM

To: Challberg, Roy C. (PS, NE)
Subject: Re: ESP-8 Information for the ESBWR

Hi Roy,

Thanks for the ESBWR information. I'll look at it Monday and see if I have any questions. One item I do need, is what is the MWe for the ESBWR? I use it and the capacity factor to normalize to the reference LWR which was 1000 MWe and 80%.

Have a good weekend.

Bob

Robert L. Nitschke
Science Fellow
INEEL, IRC 602/242
P.O. Box 1625
Idaho Falls, ID 83415-2209
Phone 208 526-1463 Fax 208 526-0690

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com> on 03/17/2003 12:39:38 PM

To: RLN@inel.gov
cc:

Subject: RE: ESP-8 Information for the ESBWR

Bob-
You're exactly right. The bounding decay heat value for one of the reload cores for ABWR (4300 MWt) was 2.9 kW. That was the decay heat after 5 years (with a 2 sigma uncertainty). This will bound the 4000 MWt ESBWR fuel.
Roy

-----Original Message-----
From: RLN@inel.gov [mailto:RLN@inel.gov]
Sent: Monday, March 17, 2003 9:09 AM
To: Challberg, Roy C. (PS, NE)
Subject: RE: ESP-8 Information for the ESBWR

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If you would please check into this for the ESBWR.

Thank you.

Bob

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com> 03/17/2003 09:43 AM

To: RLN@inel.gov
cc:
Fax to:
Subject: RE: ESP-8 Information for the ESBWR

Bob-
Our recent heat balance of our total plant design for a typical site gives us 1390 MWe, which of course is highly dependent upon site conditions and type of heat sink.
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Roy

GE Nuclear Energy
Advanced Reactor Projects
(408) 925-3317

-----Original Message-----
From: RLN@inel.gov [mailto:RLN@inel.gov]
Sent: Friday, March 14, 2003 3:19 PM
To: Challberg, Roy C. (PS, NE)
Subject: Re: ESP-8 Information for the ESBWR

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Thanks for the ESBWR information. I'll look at it Monday and see if I have any questions. One item I do need, is what is the MWe for the ESBWR? I use it and the capacity factor to normalize to the reference LWR which was 1000 MWe and 80%.

Have a good weekend.

Bob

Robert L. Nitschke
Science Fellow
INEEL, IRC 602/242
P.O. Box 1625
Idaho Falls, ID 83415-2209
Phone 208 526-1463 Fax 208 526-0690

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com> on 03/17/2003 09:43:07 AM

To: RLN@inel.gov
cc:

Subject: RE: ESP-8 Information for the ESBWR .

Bob-
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Let me know if you need anything else.
Roy

GE Nuclear Energy
Advanced Reactor Projects
(408) 925-3317

-----Original Message-----
From: RLN@inel.gov [mailto:RLN@inel.gov]
Sent: Friday, March 14, 2003 3:19 PM
To: Challberg, Roy C. (PS, NE)
Subject: Re: ESP-8 Information for the ESBWR

Hi Roy,

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Have a good weekend.

Bob

Robert L. Nitschke
Science Fellow
INEEL, IRC 602/242
P.O. Box 1625
Idaho Falls, ID 83415-2209
Phone 208 526-1463 Fax 208 526-0690

"Challberg, Roy C. (PS, NE)" <roy.challberg@gene.GE.com> on 03/14/2003 02:55:12 PM

To: "Nitschke, Robert (INEEL)" <rln@inel.gov>
cc: "Cambria, Michael (Parsons)" <Michael.Cambria@parsons.com>, "Mundy, Thomas (Exelon)" <thomas.mundy@exeloncorp.com>, "Rao, Atambir" <atambir.rao@gene.GE.com>

Subject: ESP-8 Information for the ESBWR

Robert-
Attached is a file describing the GE ESBWR fuel and core, in response to the ESP-8 questionaire (S-3/4 information).

If you need further information, please don't hesitate to contact me directly.

Roy Challberg
GE Nuclear Energy
Advanced Reactor Projects
(408) 925-3317
<mailto:roy.challberg@gene.ge.com>

<<S3_S4 Questions-ESBWR.doc>>

S3_S4 Questions-ESBWR.do

ESP 8: Reactor Vendor Questionnaire

Information on Annual Fuel Requirements

1. Define Standard Technical Configuration.
   - Provide expected reactor power, MW_t and MW_e for each reactor

     For the GE ABWR and ESBWR the uprated thermal power of the ABWR of 4300 MWt is used to bound both reactors. For reference the currently certified power level of the ABWR is 3926 MWt and the ESBWR design value is 4000 MWt

   - Number of modules or reactors expected for a typical unit configuration for small modular systems

     The ABWR and ESBWR are both designed as **single unit plants**.

2. Expected Fuel Loading
   - Provide Initial Core Fuel Loading in MTU

     The initial core load for the ABWR and ESBWR are approximately equal. For the ABWR the initial core load is 156.96 MT of Uranium.
     ABWR ~ 157 MTU
     ESBWR ~ 157 MTU

   - Provide Annual Average Fuel Loading in MTU based on 40 years of operation

     The average annual fuel loading in MTU is 32.76. This is based on an average capacity factor of 95%. This capacity factor is subject to variation by the operating practices of the utility but in GE's view represents a reasonably achievable measure. This figure includes allowances for refueling and maintenance outages but does not include any provision for extended outages.

   [Note: Provide the basis for the above estimates, i.e. estimated unit capacity factor, refueling/maintenance outage frequencies and durations, and average expected energy produced per year.]

3. Average Fuel Enrichment in % U-235
   [Note: Provide table of MTU and enrichment if multiple fuel enrichments are normally used for the initial core or fuel reloads]

   The batch average enrichment of the core is less than 3.5% for the initial core and less than 4.5% for the subsequent reloads.

4. Fuel form
   - Provide Fuel Assembly (or Basic Fuel Unit) Drawing - See Figure 1

- Provide a Table giving the following for each fuel unit:

- An estimate of the typical number of fuel assemblies or units required for the initial core and the average expected number of fuel assemblies or units per year for core reloads

Table 1

Parameter	ABWR	ESBWR
Number of bundles in core	872	1020
Active fuel length	381 cm	305 cm
Fuel bundle average mass (with channel)	298 kg	238 kg
Fuel bundle average mass (w/o channel)	266 kg	213 kg
Bundle average Uranium mass	180 kg	144 kg
Bundle outside dimensions	14.2 cm X 14.2 cm	14.2 cm X 14.2 cm
Bundle overall length	447 cm	378 cm
Mass of Zircaloy (per bundle)	~85 kg	~68 kg
Mass of Stainless Steel (per bundle)	~6.8 kg	~6.8 kg
Mass of Inconel (per bundle)	~0.5 kg	~0.5 kg

See Table 1 above. The information provided in response to this request is based on the GE-14 fuel type, which is the latest offering of the GE fuels group. The basic design of the GE-14 is the same as earlier BWR fuels offered by GE as far as overall dimensions. However, improvements in the design have been made to optimize the fuel utilization.

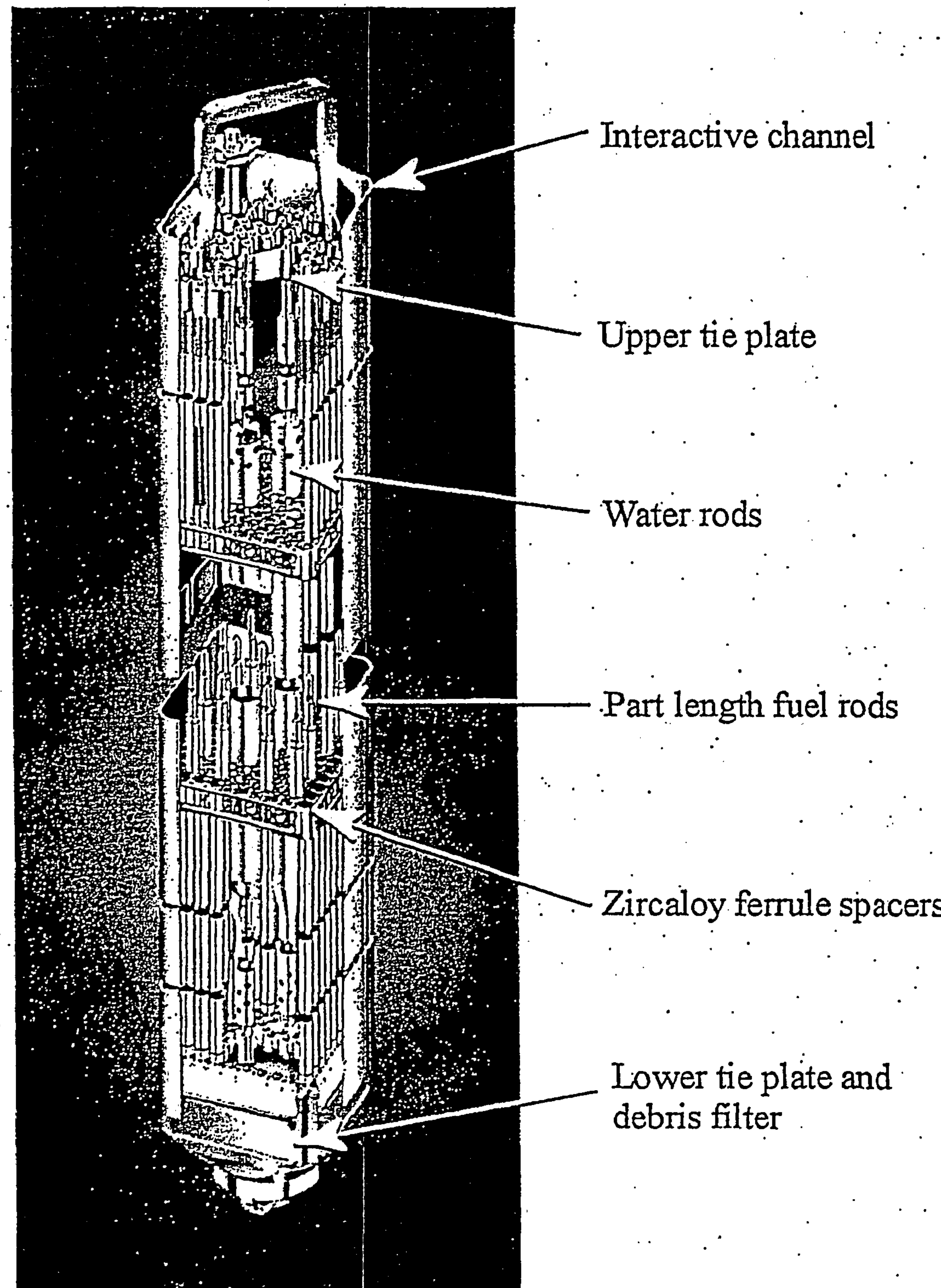
The cross sectional area of the ESBWR bundles will be the same as the ABWR. However they are approximately 15% shorter, but with more bundles in the core, so the net effect is that approximately the same amount of Uranium is held in the core as for the ABWR.

5. Fuel materials
    - Provide a table of fuel material types and mass for a typical fuel unit including a description of fuel, structural, and cladding materials

See Table 1 above. The channel, fuel rods (cladding), water rods, spacers and end plugs are all fabricated from Zircaloy.

The upper and lower tie plates and assorted fasteners are fabricated from stainless steel.

There are a number of small components that are fabricated from inconel.



GE 14 Fuel Bundle
Figure 1

6. Define the expected typical transport mode (i.e. truck, rail, etc.) for delivery of the unirradiated fuel from the fabrication facility or port of entry to the reactor site

   Typical shipment of new fuel from the GE fuel facility in Wilmington, NC is by flat bed tractor trailer.

7. Provide a general description of the transport containers expected to be used for delivery of unirradiated fuel
   - Capacity of each container, i.e. number of fuel units per container

     The transport containers consist of a dual packaging system. Two fuel assemblies are first packed in a padded steel box. The steel box is then packaged inside a padded wooden crate. The dimensions of a typical wooden crate are 30" x 30" x 15'6" (ESBWR fuel assemblies crate would be ~ 13'6" long)

   - Number of containers that can be transported on one legal weight truck shipment

     The number of ABWR bundles that typically can be shipped on a single truck is either 28 or 30 and is limited by weight. The single truck could carry up to 36 ESBWR bundles based on the lighter weight.

   [Note: This data is intended to allow for a determination of the number of shipments and MTU for the initial core loading and the average number of shipments and MTU per year for core reloads.]

Information on Expected Low Level Waste Production

1. Estimated annual average LLW production expected from reactor operations
   - Provide an estimate of the expected volumes and curies of LLW

     The production of LLW is in large part controlled by the practices of the owner. GE in the design certification chose not to establish unreasonable expectations for future owners and as such followed the maximum target values. In this case the volume of LLW is 100 cubic meters per year with an estimated curie content of 2700 Ci.

2. LLW expected from reactor decontamination and decommissioning
   - Provide an estimate of the expected volumes and curies of LLW produced due to reactor decontamination and decommissioning

     The process for decontamination and decommissioning of an ABWR is outside the control of GE and subject to a great deal of variation depending on the timing and the methods chosen. As such, GE is unable to provide a reasonable estimate for these values.

Information on Spent Fuel Production/Transport

1. Spent Fuel Shipments
   - Provide an estimate of the quantity (MTU) of irradiated fuel that can be transported in one legal weight truck cask [25 ton cask] or typical rail cask [100 ton cask], assuming 5 year cooling after discharge.
     [Note: Estimate should be in MTU (based on unirradiated MTU) and number of fuel units to allow for a determination of the average number of spent fuel shipments expected per year of reactor operation.]

   GE is not familiar with the constraints of fuel assemblies that can be transported in the commercially available casks. What we can tell you is the GE-14 fuel type is nearly identical to the other GE fuel types and the number of BWR fuel assemblies that can be shipped should not be different for the ABWR.

   If a "standard" size BWR spent fuel cask were used for spent ESBWR fuel, less fuel per cask shipment would result. With the shorter fuel assemblies it would be reasonable to expect a new cask design would be possible. Assuming the cask load or quantity based on either curie content or total decay heat, a larger cask could be conceived and therefore more ESBWR spent fuel bundles could be shipped per cask.

2. Provide the average fuel burnup in MWd/MTU

   After achieving an equilibrium core, the batch average burnup is 46 GWd/MT

3. Provide an estimate of the decay heat in watts per MTU after 5 years of decay from fuel discharge

   The estimated decay heat per MTU, 5 years after discharge from the core is between 18-22 kilowatts.

4. Provide estimates of the spent fuel inventories and radioactivity, in Ci per MTU, after 5 years of decay
   - Fission product inventory
   - Actinide inventory
   - Total radioactivity
   - Krypton-85 inventory

   See Table 2 for ABWR determination. This table is based on 4300 MWt ABWR (power uprated).

The fuel type is GE 14 with the following characteristics.

	Initial Core	Reload 1	Reload 2 to Eq.
Core Size, number of bundles	872	872	872
Core Thermal Power, MWth	4300	4300	4300
Operating Cycle Length, days	605	605	605
Operating Capacity Factor, %	100	100	100
Refueling Outage Duration, days	30	30	30
Refueling Interval, months	21	21	21
Loaded Batch Size	872	240	316
Batch Average Enrichment, w/o U235	3.5	4.5	4.5
Average Bundle Mass, KgU	180	180	180
Batch Average Burnup, GWd/MT	36	46	46

This particular analysis should bound the 4000 MWt ESBWR core.

[Note: If available, please provide a complete set of the ORIGEN run results (or other applicable code for the appropriate reactor type) detailing the spent fuel inventories at 5 years decay to answer questions 3 and 4.]

Table 2

Activation Products		Actinides + Daughters		Fission Products	
Isotope	Curie/MTU	Isotope	Curie/MTU	Isotope	Curie/MTU
Ag-109m	7.76E-04	Am-241	1.34E+03	Ag-108	3.44E-06
Ar-37	5.48E-16	Am-242	3.32E+01	Ag-109m	1.36E-04
Ar-39	3.42E-04	Am-242m	3.34E+01	Ag-110	4.44E-01
C-14	7.70E-01	Am-243	3.24E+01	Ag-110m	3.34E+01
Ca-41	1.18E-03	Am-245	2.50E-09	Ba-137m	1.18E+05
Ca-45	8.65E-04	Bi-212	4.49E-02	Cd-113m	6.13E+01
Cd-109	7.76E-04	Bk-249	1.72E-04	Cd-115m	6.79E-10
Cd-115m	1.12E-13	Cm-241	1.01E-17	Ce-141	1.49E-11
Cl-36	1.86E-02	Cm-242	5.51E+01	Ce-144	1.14E+04
Co-58	7.49E-05	Cm-243	3.69E+01	Cs-134	4.81E+04
Co-60	2.73E+03	Cm-244	4.86E+03	Cs-135	8.22E-01
Cr-51	5.28E-16	Cm-245	6.56E-01	Cs-137	1.24E+05
Eu-152	1.08E-03	Cm-246	1.41E-01	Eu-152	1.09E+01
Eu-154	1.53E+02	Np-235	5.01E-04	Eu-154	1.01E+04
Eu-155	7.14E+01	Np-237	6.16E-01	Eu-155	5.22E+03
Fe-55	3.35E+03	Np-238	1.67E-01	Gd-153	1.41E-01
Fe-59	4.32E-10	Np-239	3.24E+01	H-3	5.34E+02
Gd-153	2.26E+01	Np-240m	9.23E-07	I-129	4.20E-02
H-3	5.24E-04	Pa-233	6.16E-01	In-114	4.21E-11
Hf-175	3.05E-07	Pa-234m	3.13E-01	In-114m	4.39E-11
Hf-181	1.08E-10	Pb-212	4.49E-02	In-115m	4.77E-14
Ho-166m	2.39E-02	Po-212	2.88E-02	Kr-85	8.90E+03
In-113m	1.95E-02	Po-216	4.49E-02	Nb-93m	7.54E-01

Activation Products

Isotope	Curie/MTU
In-114	1.63E-09
In-114m	1.70E-09
Ir-192	8.59E-08
K-42	1.63E-12
Lu-177	9.04E-07
Lu-177m	3.93E-06
Mn-54	3.46E+01
Mo-93	1.95E-02
Nb-93m	1.98E-01
Nb-94	1.76E-01
Nb-95	4.55E-04
Nb-95m	1.52E-06
Ni-59	2.59E+00
Ni-63	4.20E+02
P-32	2.89E-08
Re-188	9.56E-08
Ru-103	8.38E-16
S-35	1.84E-05
Sb-124	2.93E-08
Sb-125	9.16E+02
Sc-46	7.04E-07
Sn-113	1.95E-02
Sn-119m	5.20E+01
Sn-121m	1.12E+00
Sn-123	2.40E-02
Sr-89	8.94E-10
Sr-90	6.68E-03
Ta-182	1.74E-01
Tb-160	1.49E-03
Tc-99	4.80E-03
Te-123m	2.45E-04
Te-125m	2.24E+02
Te-127	7.52E-07
Te-127m	7.67E-07
Tm-170	2.43E-07
W-181	1.82E-04
W-185	1.08E-05
W-188	9.46E-08
Y-90	6.68E-03
Y-91	6.98E-08
Zn-65	2.39E-03
Zr-93	6.86E-01
Zr-95	2.05E-04
Total	7.98E+03

Actinides + Daughters

Isotope	Curie/MTU
Pu-236	3.56E-01
Pu-237	4.37E-12
Pu-238	6.14E+03
Pu-239	3.87E+02
Pu-240	6.15E+02
Pu-241	1.22E+05
Pu-242	2.24E+00
Pu-243	5.85E-07
Ra-224	4.49E-02
Rn-220	4.49E-02
Th-228	4.49E-02
Th-231	2.20E-02
Th-234	3.13E-01
Tl-208	1.61E-02
U-232	6.00E-02
U-234	1.47E+00
U-235	2.20E-02
U-236	3.77E-01
U-237	3.00E+00
U-238	3.13E-01
U-240	9.23E-07
Total	1.36E+05

Fission Products

Isotope	Curie/MTU
Nb-95	6.78E-03
Nb-95m	2.27E-05
Pd-107	1.46E-01
Pm-146	1.84E+00
Pm-147	3.37E+04
Pm-148	8.59E-11
Pm-148m	1.52E-09
Pr-144	1.14E+04
Pr-144m	1.37E+02
Rh-102	4.67E-01
Rh-103m	1.09E-08
Rh-106	1.64E+04
Ru-103	1.21E-08
Ru-106	1.64E+04
Sb-124	1.17E-06
Sb-125	4.45E+03
Sb-126	1.43E-01
Sb-126m	1.02E+00
Se-79	5.61E-01
Sm-151	5.60E+02
Sn-119m	1.24E+00
Sn-121m	2.46E-01
Sn-123	1.60E-01
Sn-126	1.02E+00
Sr-89	8.33E-06
Sr-90	8.85E+04
Tb-160	3.14E-05
Tc-99	1.74E+01
Te-123m	6.79E-04
Te-125m	1.09E+03
Te-127	1.02E-01
Te-127m	1.04E-01
Te-129	1.02E-12
Te-129m	1.57E-12
Y-90	8.85E+04
Y-91	3.38E-04
Zr-93	2.50E+00
Zr-95	3.05E-03
Total	5.87E+05

Activation Products

Isotope	Curie/MTU
Ag	7.78E-04
Ar	3.42E-04
Be	2.12E-06
C	7.70E-01
Ca	2.05E-03
Cd	7.76E-04
Cl	1.86E-02
Co	2.73E+03
Cr	5.28E-16
Eu	2.24E+02
Fe	3.35E+03
Gd	2.26E+01
H	5.24E-04
Hf	1.01E-06
Ho	2.39E-02
I	1.75E-13
In	1.95E-02
Ir	1.70E-07
K	3.20E-08
Lu	4.84E-06
Mn	3.46E+01
Mo	1.95E-02
Nb	3.75E-01
Ni	4.23E+02
Os	7.29E-10
P	2.89E-08
Pb	8.97E-08
Re	2.45E-07
Ru	2.66E-14
S	1.84E-05
Sb	9.16E+02
Sc	7.04E-07
Si	2.88E-08
Sm	4.19E-06
Sn	5.32E+01
Sr	6.68E-03
Ta	1.74E-01
Tb	1.49E-03
Tc	4.80E-03
Te	2.24E+02
Tm	7.28E-06
V	2.01E-14
W	1.93E-04

Actinides + Daughters

Isotope	Curie/MTU
Am	1.44E+03
Bi	4.49E-02
Bk	1.72E-04
Cm	4.95E+03
Np	3.32E+01
Pa	9.30E-01
Pb	4.49E-02
Po	7.36E-02
Pu	1.29E+05
Ra	4.49E-02
Rn	4.49E-02
Th	3.80E-01
Tl	1.61E-02
U	5.24E+00
Total	1.36E+05

Fission Products

Isotope	Curie/MTU
Ag	3.38E+01
Ba	1.18E+05
Cd	6.13E+01
Ce	1.14E+04
Cs	1.72E+05
Eu	1.53E+04
Gd	1.41E-01
H	5.34E+02
Ho	4.51E-03
I	4.20E-02
In	9.98E-11
Kr	8.90E+03
La	1.47E-10
Nb	7.61E-01
Nd	2.17E-09
Pd	1.46E-01
Pm	3.37E+04
Pr	1.15E+04
Rb	2.99E-05
Rh	1.64E+04
Ru	1.64E+04
Sb	4.45E+03
Se	5.61E-01
Sm	5.60E+02
Sn	2.66E+00
Sr	8.85E+04
Tb	3.14E-05
Tc	1.74E+01
Te	1.09E+03
Xe	6.58E-17
Y	8.85E+04
Zr	2.50E+00
Total	5.87E+05

Activation Products		Actinides + Daughters		Fission Products	
Isotope	Curie/MTU	Isotope	Curie/MTU	Isotope	Curie/MTU
Y	6.68E-03				
Zn	2.39E-03				
Zr	6.86E-01				
Total	7.98E+03				

AP-1000

"Winters, James W."
<winterjw@westinghouse.com>

02/06/2003 01:26 PM

To: "Winters, James W." <winterjw@westinghouse.com>, "'Cambria, Michael'" <Michael.Cambria@parsons.com>
cc: "Tom Mundy (E-mail)" <thomas.mundy@exeloncorp.com>, "Robert L. Nitschke (E-mail)" <rln@inel.gov>, "Wayne Schofield (E-mail)" <Wschofie@ch2m.com>
Fax to:
Subject: RE: Data request for early site permit applications

Item #1:

Sizewell's decommissioning plan says they will generate about 13,000 TBq (350,000 Ci) from decommissioning. We estimate on the order of 50% of that for AP1000, or about 200,000 Ci.

We will include this in the siting guide with a note something like "Estimated based upon Sizewell B's estimate of 13,000 TBq."

Item #2:

60000 MWD/MTU is the current peak rod burnup limit (actually it is 62000 MWD/MTU).

21000 MWD/MTU is the approximate cycle burnup for an 18 Month (520 EFPD) Equilibrium Cycle.

48700 MWD/MTU is the approximate region average discharge burnup for each feed region (68 Assemblies) assuming continuous 18 Month (520 EFPD) Equilibrium Cycles.

Jim

> -----------
> From: Cambria, Michael[SMTP:Michael.Cambria@parsons.com]
> Sent: Friday, January 24, 2003 2:52 PM
> To: 'Winters, James W.'
> Cc: Tom Mundy (E-mail); Robert L. Nitschke (E-mail); Wayne
Schofield
> (E-mail)
> Subject: RE: Data request for early site permit applications
>
> Jim:
>
> Just a note to inquire on how you are making out with generating the
> requested info. If you could let me know if and when we might expect it,
> it
> would be helpful.
>
> Thank you
>
> Mike
>
> -----Original Message-----
> From: Winters, James W. [mailto:winterjw@westinghouse.com]
> Sent: Friday, January 17, 2003 12:48 PM
> To: 'Winters, James W.'; 'Cambria, Michael'
> Cc: 'Cummins, Ed'; 'Demetri, Kathryn J.'; 'Grant, Eddie R.'; 'George
> Zinke (E-mail)'; 'Marvin Smith (E-mail)'; 'Meneely, Timothy K.'; 'Steve
> Routh (E-mail)'; 'Spencer Semmes (E-mail)'; 'Mundy, Thomas P.'; 'Vijuk,
> Ronald P.'; 'Wayne Schofield (E-mail)'; 'RLN(a)inel.gov'; Ioannidi, John
> Subject: RE: Data request for early site permit applications
>

>
> Kathy,
>
> Please take the lead on this and then send me the information. I will get
> it to Michael and include it in our siting guide.
>
> Jim
>
> > -----------
> > From: Cambria, Michael[SMTP:Michael.Cambria@parsons.com]
> > Sent: Thursday, January 16, 2003 1:11 PM
> > To: 'Winters, James W.'
> > Cc: 'Cummins, Ed'; 'Demetri, Kathryn J.'; 'Grant, Eddie R.';
> 'George
> > Zinke (E-mail)'; 'Marvin Smith (E-mail)'; 'Meneely, Timothy K.'; 'Steve
> > Routh (E-mail)'; 'Spencer Semmes (E-mail)'; 'Mundy, Thomas P.'; 'Vijuk,
> > Ronald P.'; 'Wayne Schofield (E-mail)'; 'RLN(a)inel.gov'; Ioannidi, John
> > Subject: RE: Data request for early site permit applications
> >
> > Jim:
> >
> > After reviewing your input the following is some outstanding data needs
> > and/or questions:
> >
> > 1) Need your curie estimate for the D&D; and
> >
> > 2) Please clarify the burnup #? Presently it is stated the design
> burnup
> > is
> > 60,000 MWd/MTU while the expected is 21,000. What is the average fuel
> > burnup over the 40 year
> > operational period?
> >
> > If you have any questions on the above please contact Bob Nitschke of
> > INEEL
> > at (208) 526-1463 or by email at rln@inel.gov.
> >
> > Thanks
> >
> > Mike
> >
> >
> > -----Original Message-----
> > From: Winters, James W.
> > Sent: Thursday, December 19, 2002 3:58 PM
> > To: Winters, James W.; 'RLN(a)inel.gov'
> > Cc: Cummins, Ed; Demetri, Kathryn J.; 'Grant, Eddie R.'; George Zinke
> > (E-mail); Marvin Smith (E-mail); Meneely, Timothy K.; Steve Routh
> > (E-mail); Spencer Semmes (E-mail); 'Mundy, Thomas P.'; Vijuk, Ronald P.;
> > Winters, James W.; Wayne Schofield (E-mail); Meneely, Timothy K.;
> > Cambria, Michael
> > Subject: RE: Data request for early site permit applications
> >
> >
> > Here are our revised responses to the ESP-8 questions. We have included
> > the
> > page 3 items as Items 10 through 13 in the fuel information section.
> This
> > information will also be added to Revision 3 of our siting guide. The
> > information requested for spent fuel shipments is not available right
> now

> > since the cognizant engineer is on holiday. We will send it as soon as
> he
> > is back (1/2). The proper radwaste value is 1830 curies per year
> > corresponding to the DCD. This will also be corrected in Revision 3 of
> > the
> > Siting Guide.
> >
> > <<Responses to ESP 8 R2.doc>>
> >
> > Item 13 references an AP1000 calculation note for the ORIGEN data
> related
> > to
> > fuel inventories and radioactivity. Attached below are the relevant
> > tables
> > associated with that calculation.
> >
> > <<AP1000 SF Curie.pdf>>
> >
> > Jim
> >
> > > ----------
> > > From: RLN@inel.gov[SMTP:RLN@inel.gov]
> > > Sent: Monday, December 16, 2002 1:17 PM
> > > To: Winters, James W.
> > > Cc: Cummins, Ed; Demetri, Kathryn J.; 'Grant, Eddie R.'; George
> > Zinke
> > > (E-mail); Marvin Smith (E-mail); Meneely, Timothy K.; 'Michael
> Cambria';
> > > Steve Routh (E-mail); Spencer Semmes (E-mail); 'Mundy, Thomas P.';
> > Vijuk,
> > > Ronald P.; Winters, James W.; Wayne Schofield (E-mail)
> > > Subject: RE: Data request for early site permit applications
> > >
> > > <<File: Responses to ESP 8 R1.doc>>
> > >
> > > Hi Jim,
> > >
> > > Thanks for your response. I am not sure why you did not receive a
> page
> > 3.
> > > It should have looked something like this:
> > >
> > > Information on Spent Fuel Production/Transport
> > >
> > > 1. Spent Fuel Shipments
> > > * Provide an estimate of the quantity (MTU) of irradiated fuel
> that
> > > can
> > > be transported in one legal weight truck cask [25 ton cask] or typical
> > > rail
> > > cask [100 ton cask], assuming 5 year cooling after discharge.
> > > [Note: Estimate should be in MTU (based on unirradiated MTU) and
> > > number of fuel units to allow for a determination of average number of
> > > spent fuel shipments per year of reactor operation.]
> > > 2. Provide the average fuel burnup in MWd/MTU
> > > 3. Provide an estimate of the decay heat in watts per MTU after 5
> years
> > > of
> > > decay from fuel discharge
> > > 4. Provide estimates of the spent fuel inventories and radioactivity,
> > in

> > > Ci per MTU, after 5 years of decay
> > > * Fission product inventory
> > > * Actinide inventory
> > > * Total radioactivity
> > > * Krypton-85 inventory
> > > [Note: If available, please provide a complete set of ORIGEN run
> results
> > > (or other applicable code for the applicable reactor type) detailing
> the
> > > spent fuel inventories at 5 years decay to answer questions 3 and 4.]
> > >
> > > -------
> > >
> > > As such, we will still need information on the number and types of
> spent
> > > fuel shipment, average burnup, decay heat, etc.
> > >
> > >
> > > Also if I may, one question on your latest submittal. In the
> attachment
> > > "Responses to ESP 8 R1", it states 1830 curies per year of solid
> waste.
> > > The AP1000 Siting Guide document on pages 33 and 36 show 1100 curies
> per
> > > year.
> > >
> > > Thank you.
> > >
> > > Bob
> > >
> > > phone 208.526-1463
> > >
> > >
> > >
> > >
> > > "Winters, James
> > >
> > > W." To: "Winters,
> James
> > > W."
> > > <winterjw@westing
> > > <winterjw@westinghouse.com>, "'Michael Cambria'"
> > > house.com>
> > > <Michael.Cambria@parsons.com>
> > > cc: "'Mundy,
> Thomas
> > > P.'"
> > > 12/13/2002 09:35
> > > <thomas.mundy@exeloncorp.com>, "Cummins, Ed"
> > > AM
> > > <cumminwe@westinghouse.com>, "Vijuk, Ronald P."
> > >
> > > <vijukrp@westinghouse.com>, "'Grant, Eddie R.'"
> > >
> > > <eddie.grant@exeloncorp.com>, "Marvin Smith (E-mail)"
> > >
> <Marvin_Smith@dom.com>,
> > > "Robert L. Nitschke (E-mail)"
> > > <rln@inel.gov>,
> "Spencer
> > > Semmes (E-mail)"

> > >
> > <Spencer_Semmes@dom.com>,
> > > "Steve Routh (E-mail)"
> > > <sdrouth@bechtel.com>,
> > > "Wayne Schofield (E-mail)"
> > > <Wschofie@ch2m.com>,
> > > "George Zinke (E-mail)"
> > > <GZINKE@entergy.com>,
> > > "Vijuk, Ronald P."
> > >
> > > <vijukrp@westinghouse.com>, "Demetri, Kathryn J."
> > >
> > > <demetrkj@westinghouse.com>, "Meneely, Timothy K."
> > >
> > > <meneeltk@westinghouse.com>
> > > Fax to:
> > >
> > > Subject: RE: Data
> > request
> > > for early site permit
> > > applications
> > >
> > >
> > >
> > >
> > >
> > >
> > >
> > >
> > > With a little help from my friends, here are the answers to your
> > > questions.
> > > These also cover the email you sent me later in the day on the 9th.
> For
> > > the
> > > record, we never received the page 3 Bob talks about, so I hope you
> can
> > > apply this information to your page 3.
> > >
> > > Fuel reload data:
> > > * Cycle Length - 18 months - 520 EFPD @ 3400 MWT
> > > * Capacity Factor - 95% including refueling outage
> > > * Reload fuel requirement - 68 Fuel Assemblies
> > > * Average Enrichment - 4.51 w/o U235
> > >
> > > Spent fuel data:
> > > > * At 5 years decay, the average spent fuel assembly curie
> > > content:
> > > > Actinides 8.506E+04 curies
> > > > Fission Products 4.450E+05 curies
> > > > Total 5.301E+05
> curies
> > > >
> > > LLW from Decommissioning:
> > > * No AP1000 specific estimate has been made. Information
> > from
> > > Sizewell indicates 6200 cubic meters of LLW from decommissioning. The
> > > AP1000 value should be significantly less (maybe half) considering the
> > > design differences.
> > >
> > >
> > > I have also incorporated this information into our response to ESP 8
> > > document.

> > >
> > > <<Responses to ESP 8 R1.doc>>
> > >
> > > This information is also being added to our Siting parameters
> document.
> > > Thanks for your interest.
> > >
> > > Jim
> > > 413-374-5290
> > >
> > > > ----------
> > > > From:		Michael Cambria[SMTP:Michael.Cambria@parsons.com]
> > > > Sent:		Monday, December 09, 2002 12:14 PM
> > > > To:		'Winters, James W.'
> > > > Cc:		'Mundy, Thomas P.'; 'Cummins, Ed'; 'Vijuk, Ronald
> > P.';
> > > 'Grant, Eddie
> > > > R.'; Marvin Smith (E-mail); Robert L. Nitschke (E-mail); Spencer
> > Semmes
> > > > (E-mail); Steve Routh (E-mail); Wayne Schofield (E-mail); George
> Zinke
> > > > (E-mail)
> > > > Subject:	RE: Data request for early site permit applications
> > > >
> > > > Jim:
> > > >
> > > > I want to thank you for your input to our ESP 8 Questionnaire.
> After
> > > > reviewing the information provided by you there is some additional
> > data
> > > > that is needed by us to complete our assessment. The main data items
> > > that
> > > > are missing from your response are the average enrichment for the
> > reload
> > > > fuel along with the expected average capacity factor and information
> > on
> > > > the curies contained in the spent fuel at 5 years after discharge.
> > The
> > > > first two items are needed to calculate the fuel requirements on an
> > > > average annual basis and the information on curies contained is
> needed
> > > to
> > > > look at transport impacts.
> > > >
> > > > If you could supplement your response with this information it would
> > be
> > > > greatly appreciated. Thank you for your cooperation.
> > > >
> > > > Regards,
> > > >
> > > > Mike
> > > >
> > > > -----Original Message-----
> > > > From: Winters, James W.
> > > > Sent: Tuesday, December 03, 2002 4:52 PM
> > > > To: Winters, James W.; 'Grant, Eddie R.'
> > > > Cc: Mundy, Thomas P.; Cummins, Ed; Winters, James W.; Vijuk, Ronald
> > P.;
> > > > Cambria, Michael
> > > > Subject: RE: Data request for early site permit applications
> > > >

> > > >
> > > > Here is our response table. Formal letter will be FEDEXed today.
> > > >
> > > > <<Responses to ESP 8.doc>>
> > > >
> > > > Jim
> > > >
> > > > > ----------
> > > > > From: Grant, Eddie R.[SMTP:eddie.grant@exeloncorp.com]
> > > > > Sent: Monday, November 11, 2002 2:37 PM
> > > > > To: James W. Winters (E-mail)
> > > > > Cc: Michael Cambria (E-mail); Mundy, Thomas P.
> > > > > Subject: Data request for early site permit
> applications
> > > > >
> > > > > <<File: ESP-08, info request, AP1K.pdf>>
> > > > > As you are aware, Exelon Corporation, Dominion Resources Services,
> > and
> > > > > Entergy Nuclear Potomac are currently developing Early Site Permit
> > > (ESP)
> > > > > applications to facilitate the future deployment of advance
> reactor
> > > > design
> > > > > concepts. The attached letter requests some additional information
> > > > > necessary
> > > > > to complete the environmental assessment for this effort.
> > > > >
> > > > > <<ESP-08, info request, AP1K.pdf>>
> > > > >
> > > > > To meet the our schedule for submitting ESP Applications, it would
> > be
> > > > > beneficial if you could first provide existing data that is
> readily
> > > > > retrievable and then follow-up with additional data as it becomes
> > > > > available.
> > > > > Your response is requested by November 27, 2002.
> > > > >
> > > > > Thank you in advance for your cooperation in this matter.
> > > > >
> > > > > Please direct your responses to the attention of Michael J.
> Cambria
> > > at:
> > > > >
> > > > > Parsons Energy and Chemicals
> > > > > 2675 Morgantown Road
> > > > > Reading, PA 19607
> > > > > Email: michael.cambria@parsons.com
> > > <mailto:michael.cambria@parsons.com>
> > >
> > > > > (610) 855-2049
> > > > >
> > > > > Should you have any questions or require additional clarification
> > > > > regarding
> > > > > the information requested by the attached questionnaire, please
> > > contact
> > > > > Robert L. Nitschke of INEEL at (208) 526-1463 or by email at
> > > > rln@inel.gov
> > > > > <mailto:rln@inel.gov> .
> > > > >
> > > > >
> > > > > Eddie R Grant

> > > > > Exelon ESP Project
> > > > > 610-765-5001 Office
> > > > > 610-765-5545 Fax
> > > > > 850-598-9801 Cell
> > > > >
> > > > >
> > > > >
> > > > >
> > >
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> > > > > permanently delete the original and any copy of this e-mail and
> any
> > > > > printout. Thank You.
> > > > >
> > >
> ***
> > > > >
> > > > >
> > > >
> > > (See attached file: Responses to ESP 8 R1.doc)
> > >
> > >
> > >
> >
>

"Winters, James W." <winterjw@westinghouse.com> on 01/02/2003 08:00:20 AM

To: "'RLN@inel.gov'" <RLN@inel.gov>
cc: "'spencer_semmes@dom.com'" <spencer_semmes@dom.com>, "'edward.b.toll@parsons.com'" <edward.b.toll@parsons.com>, "'michael.cambria@parsons.com'" <michael.cambria@parsons.com>, "'eddie.grant@exeloncorp.com'" <eddie.grant@exeloncorp.com>, "Cummins, Ed" <cumminwe@westinghouse.com>, "Winters, James W." <winterjw@westinghouse.com>

Subject: Spent Fuel Shipping

This completes our responses to the ESP-8 questions. Currently operating plants ship spent Westinghouse fuel after 10 years after removal from the reactor. This is usually 5 years in pool and 5 years dry storage. There can be 21-28 fuel assemblies n a shipping cask and one cask per rail car. None are shipped by truck.

Jim

"Winters, James W."
<winterjw@westinghouse.com>

12/19/2002 01:57 PM

To: "Winters, James W." <winterjw@westinghouse.com>, "'RLN@inel.gov'" <RLN@inel.gov>
cc: "Cummins, Ed" <cumminwe@westinghouse.com>, "Demetri, Kathryn J." <demetrkj@westinghouse.com>, "'Grant, Eddie R.'" <eddie.grant@exeloncorp.com>, "George Zinke (E-mail)" <GZINKE@entergy.com>, "Marvin Smith (E-mail)" <Marvin_Smith@dom.com>, "Meneely, Timothy K." <meneeltk@westinghouse.com>, "'Michael Cambria'" <Michael.Cambria@parsons.com>, "Steve Routh (E-mail)" <sdrouth@bechtel.com>, "Spencer Semmes (E-mail)" <Spencer_Semmes@dom.com>, "'Mundy, Thomas P.'" <thomas.mundy@exeloncorp.com>, "Vijuk, Ronald P." <vijukrp@westinghouse.com>, "Winters, James W." <winterjw@westinghouse.com>, "Wayne Schofield (E-mail)" <Wschofie@ch2m.com>, "Meneely, Timothy K." <meneeltk@westinghouse.com>
Fax to:
Subject: RE: Data request for early site permit applications

Here are our revised responses to the ESP-8 questions. We have included the page 3 items as Items 10 through 13 in the fuel information section. This information will also be added to Revision 3 of our siting guide. The information requested for spent fuel shipments is not available right now since the cognizant engineer is on holiday. We will send it as soon as he is back (1/2). The proper radwaste value is 1830 curies per year corresponding to the DCD. This will also be corrected in Revision 3 of the Siting Guide.

<<Responses to ESP 8 R2.doc>>

Item 13 references an AP1000 calculation note for the ORIGEN data related to fuel inventories and radioactivity. Attached below are the relevant tables associated with that calculation.

<<AP1000 SF Curie.pdf>>

Jim

\> ----------
\> From: RLN@inel.gov[SMTP:RLN@inel.gov]
\> Sent: Monday, December 16, 2002 1:17 PM
\> To: Winters, James W.
\> Cc: Cummins, Ed; Demetri, Kathryn J.; 'Grant, Eddie R.'; George Zinke
\> (E-mail); Marvin Smith (E-mail); Meneely, Timothy K.; 'Michael Cambria';
\> Steve Routh (E-mail); Spencer Semmes (E-mail); 'Mundy, Thomas P.'; Vijuk,
\> Ronald P.; Winters, James W.; Wayne Schofield (E-mail)
\> Subject: RE: Data request for early site permit applications
\>
\> <<File: Responses to ESP 8 R1.doc>>
\>
\> Hi Jim,
\>
\> Thanks for your response. I am not sure why you did not receive a page 3.
\> It should have looked something like this:
\>
\> Information on Spent Fuel Production/Transport
\>
\> 1. Spent Fuel Shipments
\> * Provide an estimate of the quantity (MTU) of irradiated fuel that
\> can

> be transported in one legal weight truck cask [25 ton cask] or typical
> rail
> cask [100 ton cask], assuming 5 year cooling after discharge.
>   [Note: Estimate should be in MTU (based on unirradiated MTU) and
> number of fuel units to allow for a determination of average number of
> spent fuel shipments per year of reactor operation.]
> 2. Provide the average fuel burnup in MWd/MTU
> 3. Provide an estimate of the decay heat in watts per MTU after 5 years
> of
> decay from fuel discharge
> 4. Provide estimates of the spent fuel inventories and radioactivity, in
> Ci per MTU, after 5 years of decay
>   * Fission product inventory
>   * Actinide inventory
>   * Total radioactivity
>   * Krypton-85 inventory
> [Note: If available, please provide a complete set of ORIGEN run results
> (or other applicable code for the applicable reactor type) detailing the
> spent fuel inventories at 5 years decay to answer questions 3 and 4.]
>
> ------
>
> As such, we will still need information on the number and types of spent
> fuel shipment, average burnup, decay heat, etc.
>
>
> Also if I may, one question on your latest submittal. In the attachment
> "Responses to ESP 8 R1", it states 1830 curies per year of solid waste.
> The AP1000 Siting Guide document on pages 33 and 36 show 1100 curies per
> year.
>
> Thank you.
>
> Bob
>
> phone 208 526-1463
>
>
>
>
>   "Winters, James
>
>   W."   To: "Winters, James
> W."
>   <winterjw@westing
> <winterjw@westinghouse.com>, "'Michael Cambria'"
>   house.com>
> <Michael.Cambria@parsons.com>
>   cc: "'Mundy, Thomas
> P.'"
>   12/13/2002 09:35
> <thomas.mundy@exeloncorp.com>, "Cummins, Ed"
>   AM
> <cumminwe@westinghouse.com>, "Vijuk, Ronald P."
>
> <vijukrp@westinghouse.com>, "'Grant, Eddie R.'"
>
> <eddie.grant@exeloncorp.com>, "Marvin Smith (E-mail)"
>   <Marvin_Smith@dom.com>,
> "Robert L. Nitschke (E-mail)"
>   <rln@inel.gov>, "Spencer

> Semmes (E-mail)" <Spencer_Semmes@dom.com>,
>
> "Steve Routh (E-mail)" <sdrouth@bechtel.com>,
>
> "Wayne Schofield (E-mail)" <Wschofie@ch2m.com>,
>
> "George Zinke (E-mail)" <GZINKE@entergy.com>,
>
> "Vijuk, Ronald P."
>
> <vijukrp@westinghouse.com>, "Demetri, Kathryn J."
>
> <demetrkj@westinghouse.com>, "Meneely, Timothy K."
>
> <meneeltk@westinghouse.com>
>
> Fax to:
>
> Subject: RE: Data request
>
> for early site permit applications
>
>
>
>
>
>
>
>
> With a little help from my friends, here are the answers to your
> questions.
> These also cover the email you sent me later in the day on the 9th. For
> the
> record, we never received the page 3 Bob talks about, so I hope you can
> apply this information to your page 3.
>
> Fuel reload data:
> * Cycle Length - 18 months - 520 EFPD @ 3400 MWT
> * Capacity Factor - 95% including refueling outage
> * Reload fuel requirement - 68 Fuel Assemblies
> * Average Enrichment - 4.51 w/o U235
>
> Spent fuel data:
> > * At 5 years decay, the average spent fuel assembly curie
> content:
> > Actinides 8.506E+04 curies
> > Fission Products 4.450E+05 curies
> > Total 5.301E+05 curies
> >
> LLW from Decommissioning:
> * No AP1000 specific estimate has been made. Information from
> Sizewell indicates 6200 cubic meters of LLW from decommissioning. The
> AP1000 value should be significantly less (maybe half) considering the
> design differences.
>
>
> I have also incorporated this information into our response to ESP 8
> document.
>
> <<Responses to ESP 8 R1.doc>>
>
> This information is also being added to our Siting parameters document.
> Thanks for your interest.

>
> Jim
> 413-374-5290
>
> > ----------
> > From: Michael Cambria[SMTP:Michael.Cambria@parsons.com]
> > Sent: Monday, December 09, 2002 12:14 PM
> > To: 'Winters, James W.'
> > Cc: 'Mundy, Thomas P.'; 'Cummins, Ed'; 'Vijuk, Ronald P.';
> 'Grant, Eddie
> > R.'; Marvin Smith (E-mail); Robert L. Nitschke (E-mail); Spencer Semmes
> > (E-mail); Steve Routh (E-mail); Wayne Schofield (E-mail); George Zinke
> > (E-mail)
> > Subject: RE: Data request for early site permit applications
> >
> > Jim:
> >
> > I want to thank you for your input to our ESP 8 Questionnaire. After
> > reviewing the information provided by you there is some additional data
> > that is needed by us to complete our assessment. The main data items
> that
> > are missing from your response are the average enrichment for the reload
> > fuel along with the expected average capacity factor and information on
> > the curies contained in the spent fuel at 5 years after discharge. The
> > first two items are needed to calculate the fuel requirements on an
> > average annual basis and the information on curies contained is needed
> to
> > look at transport impacts.
> >
> > If you could supplement your response with this information it would be
> > greatly appreciated. Thank you for your cooperation.
> >
> > Regards,
> >
> > Mike
> >
> > -----Original Message-----
> > From: Winters, James W.
> > Sent: Tuesday, December 03, 2002 4:52 PM
> > To: Winters, James W.; 'Grant, Eddie R.'
> > Cc: Mundy, Thomas P.; Cummins, Ed; Winters, James W.; Vijuk, Ronald P.;
> > Cambria, Michael
> > Subject: RE: Data request for early site permit applications
> >
> >
> > Here is our response table. Formal letter will be FEDEXed today.
> >
> > <<Responses to ESP 8.doc>>
> >
> > Jim
> >
> > > ----------
> > > From: Grant, Eddie R.[SMTP:eddie.grant@exeloncorp.com]
> > > Sent: Monday, November 11, 2002 2:37 PM
> > > To: James W. Winters (E-mail)
> > > Cc: Michael Cambria (E-mail); Mundy, Thomas P.
> > > Subject: Data request for early site permit applications
> > >
> > > <<File: ESP-08, info request, AP1K.pdf>>
> > > As you are aware, Exelon Corporation, Dominion Resources Services, and
> > > Entergy Nuclear Potomac are currently developing Early Site Permit

> (ESP)
> > > applications to facilitate the future deployment of advance reactor
> > design
> > > concepts. The attached letter requests some additional information
> > > necessary
> > > to complete the environmental assessment for this effort.
> > >
> > > <<ESP-08, info request, AP1K.pdf>>
> > >
> > > To meet the our schedule for submitting ESP Applications, it would be
> > > beneficial if you could first provide existing data that is readily
> > > retrievable and then follow-up with additional data as it becomes
> > > available.
> > > Your response is requested by November 27, 2002.
> > >
> > > Thank you in advance for your cooperation in this matter.
> > >
> > > Please direct your responses to the attention of Michael J. Cambria
> at:
> > >
> > > Parsons Energy and Chemicals
> > > 2675 Morgantown Road
> > > Reading, PA 19607
> > > Email: michael.cambria@parsons.com
> <mailto:michael.cambria@parsons.com>
>
> > > (610) 855-2049
> > >
> > > Should you have any questions or require additional clarification
> > > regarding
> > > the information requested by the attached questionnaire, please
> contact
> > > Robert L. Nitschke of INEEL at (208) 526-1463 or by email at
> > rln@inel.gov
> > > <mailto:rln@inel.gov> .
> > >
> > >
> > > Eddie R Grant
> > > Exelon ESP Project
> > > 610-765-5001 Office
> > > 610-765-5545 Fax
> > > 850-598-9801 Cell
> > >
> > >
> > >
> > >
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> > > printout. Thank You.
> > >
> **
> > >
> > >
> >
> (See attached file: Responses to ESP 8 R1.doc)
>
>
>

Responses to ESP 8 R2AP1000 SF Curie.

Responses to ESP 8: Reactor Vendor Questionnaire
Revision 2

Information on Annual Fuel Requirements

1. Standard Technical Configuration

Reactor Power	3400 MW_t
Plant Power	1117 – 1150 MW_e
Number of Plants per Unit	1

2. Expected Fuel Loading

Initial Core Fuel Loading	84.5 MTU
Annual Average Fuel Loading	24.4 MTU

3. Average Fuel Enrichment (initial load)

Region 1	2.35 weight % U-235
Region 2	3.40 weight % U-235
Region 3	4.45 weight % U-235

4. Fuel Form

Fuel Assembly Drawing	See attached figure
Total mass	1730 lb/assembly
Uranium mass	0.5383 MTU/assembly
Volume (FA envelope)	13404.3 in^3
Outside Dimensions	8.426x8.426x188.8 in
Number of Assemblies (Initial)	157
Number of Assemblies (Reload)	68 on 18 month cycle

5. Fuel Materials

Fuel	211,588 lb UO_2
Structure and Cladding	43,105 lb Zircaloy or ZIRLO™
	270 lb Alloy 718 (top & bottom Grids for 157 assemblies)

6. Expected Typical Transport — Truck

7. <u>New Fuel</u> Transport Containers

Capacity	2 assemblies per container
Shipping	6 containers per truck

8. Fuel reload data:

Cycle Length	18 months - 520 EFPD @ 3400 MWT

Capacity Factor	95% including refueling outage
Reload fuel requirement	68 Fuel Assemblies
Average Enrichment	4.51 w/o U235

9. Spent fuel data:

At 5 years decay, the average spent fuel assembly curie content:

Actinides	8.506E+04 curies
Fission Products	4.450E+05 curies
Total	5.301E+05 curies

10. Spent Fuel Shipping Information

Quantity of spent fuel (MTU):

Truck Cask	To be provided later
Rail Car Cask	To be provided later

11. Average Fuel Burnup

Expected :	21000 MWD/MTU (3400 MWt x 520 efpd / 84.5 MTU)
Design	60000 MWD/MTU

12. Estimate of Decay Heat in watts per MTU after 5 years of decay

While we use ORIGEN, we have not used it for decay heat calculation for AP1000. We therefore have estimated decay heat based on ANS 1979 standards, with 0 sigma margin, at five years to be 1.127E-4 watts/watt. With core power of 3400 MW and core loading of 84.5 MTU, the estimated specific decay heat for AP1000 is 4530 watts/MTU.

13. Estimates of spent fuel inventories and radioactivity

ORIGEN results for spent fuel inventories and radioactivity are addressed by AP1000 document APP-SSAR-GS2-496. This is based on one burned AP1000 assembly, decayed to 5 years. (Note that ORIGEN was run assuming a core loading of 83.6 MTU.) The 5 year decay data is in the last column (as label indicates). Also note that the inventory units are total Curies (based on 532337.6 grams for an assembly).

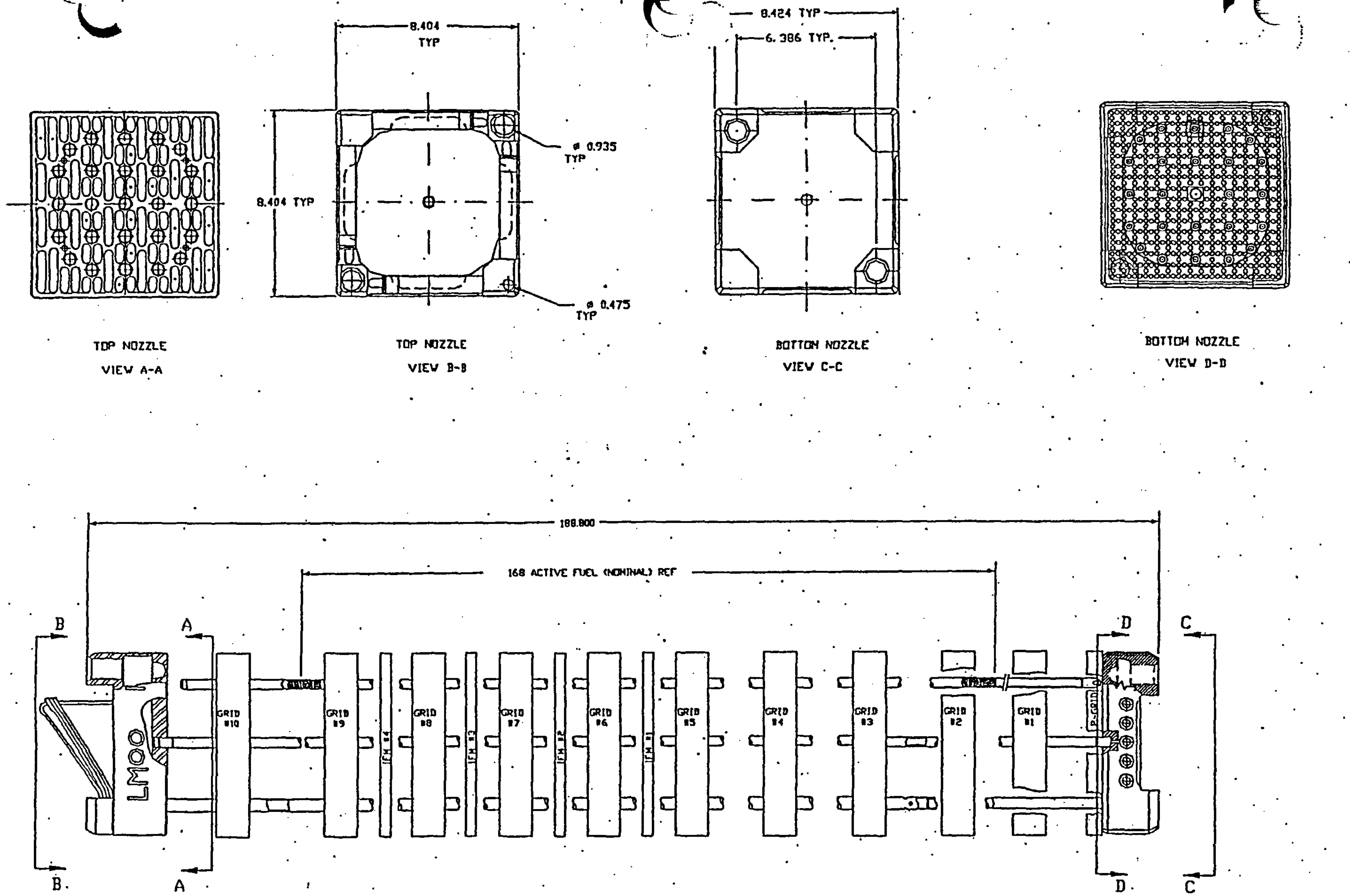
Information on Expected Low Level Waste Production

1. LLW Production

Volume	1964 cubic feet per year (average, as shipped)
Activity	1830 curies per year (average, as shipped)

2. LLW from Decommissioning

No AP1000 specific estimate has been made. Information from Sizewell indicates 6200 cubic meters of LLW from decommissioning. The AP1000 value should be significantly less (maybe half) considering the design differences.



Dimensions are in inches (nominal)

WESTINGHOUSE CONFIGURATION CONTROL
Internal Reference CN-REA-01-62 R0

Code: ORIGEN2
Version: 2.1.1
Configuration: February 3, 1995
Execution: November 30, 2001 16:51:38.57
Control Number: 5342983194974

A record of configured versions exists in the Westinghouse Engineering Technology Configuration Control Department.

1 OUTPUT UNIT = 6
PAGE 231
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

ACTINIDES+DAUGHTERS

+
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
HE 4	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	4.043E-18	4.043E-18	4.043E-18	4.043E-18	4.043E-18	4.043E-18	4.043E-18	4.043E-18	4.043E-18	4.043E-18
TL207	3.447E-08	3.440E-08	3.440E-08	3.439E-08	3.437E-08	3.390E-08	3.409E-08	3.684E-08	4.436E-08	1.254E-07
TL208	9.191E-04	9.635E-04	9.641E-04	9.279E-04	9.363E-04	1.014E-03	1.267E-03	1.660E-03	2.617E-03	1.154E-02
TL209	1.439E-08	1.441E-08	1.441E-08	1.428E-08	1.397E-08	9.592E-09	4.893E-09	4.468E-09	4.469E-09	4.541E-09
PB206	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB207	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB208	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB209	6.676E-07	6.234E-07	6.202E-07	6.611E-07	6.466E-07	4.441E-07	2.265E-07	2.069E-07	2.069E-07	2.102E-07
PB210	3.794E-09	3.800E-09	3.805E-09	3.838E-09	3.864E-09	4.005E-09	4.095E-09	4.082E-09	4.022E-09	3.693E-09
PB211	3.457E-08	3.449E-08	3.449E-08	3.448E-08	3.446E-08	3.400E-08	3.418E-08	3.695E-08	4.449E-08	1.258E-07
PB212	2.558E-03	2.558E-03	2.560E-03	2.581E-03	2.603E-03	2.823E-03	3.525E-03	4.620E-03	7.283E-03	3.212E-02
PB214	9.932E-11	9.947E-11	9.956E-11	1.002E-10	1.008E-10	1.060E-10	1.229E-10	1.512E-10	2.316E-10	2.893E-09
BI208	5.361E-18	5.361E-18	5.361E-18	5.361E-18	5.361E-18	5.361E-18	5.361E-18	5.361E-18	5.361E-18	5.360E-18
BI209	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BI210M	4.059E-18	4.059E-18	4.059E-18	4.059E-18	4.059E-18	4.059E-18	4.059E-18	4.059E-18	4.059E-18	4.059E-18
BI210	3.750E-09	3.753E-09	3.757E-09	3.780E-09	3.804E-09	3.967E-09	4.099E-09	4.084E-09	4.024E-09	3.695E-09
BI211	3.457E-08	3.449E-08	3.449E-08	3.448E-08	3.446E-08	3.400E-08	3.418E-08	3.695E-08	4.449E-08	1.258E-07
BI212	2.558E-03	2.682E-03	2.683E-03	2.583E-03	2.606E-03	2.823E-03	3.525E-03	4.620E-03	7.283E-03	3.212E-02
BI213	6.662E-07	6.671E-07	6.674E-07	6.611E-07	6.466E-07	4.441E-07	2.265E-07	2.069E-07	2.069E-07	2.102E-07
BI214	9.932E-11	9.947E-11	9.956E-11	1.002E-10	1.008E-10	1.060E-10	1.229E-10	1.512E-10	2.316E-10	2.893E-09
PO210	2.262E-09	2.266E-09	2.270E-09	2.293E-09	2.314E-09	2.486E-09	2.880E-09	3.299E-09	3.717E-09	3.697E-09

PO211M	7.287E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PO211	9.683E-11	9.658E-11	9.658E-11	9.655E-11	9.650E-11	9.519E-11	9.571E-11	1.034E-10	1.246E-10	3.522E-10
PO212	1.639E-03	1.718E-03	1.719E-03	1.655E-03	1.670E-03	1.808E-03	2.258E-03	2.960E-03	4.666E-03	2.058E-02
PO213	6.518E-07	6.527E-07	6.529E-07	6.468E-07	6.326E-07	4.345E-07	2.217E-07	2.024E-07	2.024E-07	2.057E-07
PO214	1.366E-07	1.344E-07	1.322E-07	1.190E-07	1.082E-07	5.036E-08	6.927E-09	4.902E-10	2.322E-10	2.892E-09
PO215	3.449E-08	3.449E-08	3.449E-08	3.448E-08	3.446E-08	3.400E-08	3.418E-08	3.695E-08	4.449E-08	1.258E-07
PO216	2.558E-03	2.561E-03	2.564E-03	2.586E-03	2.609E-03	2.823E-03	3.525E-03	4.620E-03	7.283E-03	3.212E-02
PO218	9.934E-11	9.949E-11	9.958E-11	1.002E-10	1.008E-10	1.060E-10	1.230E-10	1.512E-10	2.316E-10	2.893E-09
AT217	6.662E-07	6.671E-07	6.673E-07	6.611E-07	6.466E-07	4.441E-07	2.265E-07	2.069E-07	2.069E-07	2.102E-07
RN218	1.365E-07	1.343E-07	1.321E-07	1.189E-07	1.081E-07	5.025E-08	6.804E-09	3.390E-10	7.067E-13	0.000E+00
RN219	3.449E-08	3.449E-08	3.449E-08	3.448E-08	3.446E-08	3.400E-08	3.418E-08	3.695E-08	4.449E-08	1.258E-07
RN220	2.558E-03	2.561E-03	2.564E-03	2.586E-03	2.609E-03	2.823E-03	3.525E-03	4.620E-03	7.283E-03	3.212E-02
RN222	9.934E-11	9.944E-11	9.953E-11	1.002E-10	1.008E-10	1.060E-10	1.230E-10	1.512E-10	2.316E-10	2.893E-09
FR221	6.662E-07	6.671E-07	6.673E-07	6.611E-07	6.466E-07	4.441E-07	2.265E-07	2.069E-07	2.069E-07	2.102E-07
FR223	4.465E-10	4.467E-10	4.469E-10	4.483E-10	4.495E-10	4.598E-10	4.873E-10	5.306E-10	6.276E-10	1.735E-09
RA222	1.365E-07	1.343E-07	1.321E-07	1.189E-07	1.081E-07	5.025E-08	6.804E-09	3.390E-10	7.067E-13	0.000E+00
RA223	3.449E-08	3.449E-08	3.449E-08	3.448E-08	3.446E-08	3.400E-08	3.418E-08	3.695E-08	4.449E-08	1.258E-07
RA224	2.558E-03	2.561E-03	2.564E-03	2.586E-03	2.608E-03	2.823E-03	3.525E-03	4.620E-03	7.283E-03	3.212E-02
RA225	6.917E-07	6.805E-07	6.695E-07	6.057E-07	5.561E-07	3.255E-07	2.136E-07	2.066E-07	2.068E-07	2.102E-07
RA226	1.004E-10	1.005E-10	1.006E-10	1.013E-10	1.020E-10	1.074E-10	1.230E-10	1.512E-10	2.316E-10	2.893E-09
RA228	4.438E-12	4.441E-12	4.445E-12	4.468E-12	4.488E-12	4.656E-12	5.109E-12	5.827E-12	7.453E-12	2.615E-11
AC225	6.662E-07	6.669E-07	6.671E-07	6.609E-07	6.464E-07	4.440E-07	2.265E-07	2.069E-07	2.069E-07	2.102E-07
AC227	3.235E-08	3.237E-08	3.238E-08	3.248E-08	3.257E-08	3.332E-08	3.531E-08	3.845E-08	4.547E-08	1.257E-07
AC228	6.169E-07	1.588E-07	4.090E-08	1.205E-11	4.494E-12	4.656E-12	5.109E-12	5.828E-12	7.454E-12	2.615E-11
TH226	1.365E-07	1.343E-07	1.321E-07	1.189E-07	1.081E-07	5.025E-08	6.804E-09	3.390E-10	7.067E-13	0.000E+00
TH227	3.402E-08	3.402E-08	3.402E-08	3.396E-08	3.386E-08	3.319E-08	3.406E-08	3.698E-08	4.380E-08	1.241E-07
TH228	2.587E-03	2.592E-03	2.597E-03	2.626E-03	2.652E-03	2.874E-03	3.511E-03	4.603E-03	7.263E-03	3.211E-02
TH229	2.064E-07	2.064E-07	2.064E-07	2.064E-07	2.064E-07	2.064E-07	2.065E-07	2.066E-07	2.068E-07	2.102E-07
TH230	1.878E-07	1.881E-07	1.883E-07	1.899E-07	1.913E-07	2.030E-07	2.364E-07	2.946E-07	4.444E-07	3.049E-06

1. OUTPUT UNIT = 6
PAGE 232
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

+ ACTINIDES+DAUGHTERS

POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
TH231	1.329E-02	1.096E-02	9.281E-03	5.475E-03	5.009E-03	4.921E-03	4.922E-03	4.922E-03	4.922E-03	4.923E-03
TH232	2.980E-11	2.981E-11	2.983E-11	2.992E-11	3.000E-11	3.066E-11	3.238E-11	3.495E-11	4.026E-11	8.212E-11
TH233	2.971E-02	4.631E-12	7.217E-22	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TH234	1.626E-01	1.626E-01	1.626E-01	1.626E-01	1.626E-01	1.625E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01
PA231	3.967E-07	3.971E-07	3.974E-07	3.986E-07	3.995E-07	4.065E-07	4.240E-07	4.501E-07	5.034E-07	9.202E-07
PA232	1.986E-02	1.525E-02	1.170E-02	2.191E-03	4.895E-04	2.542E-09	4.164E-23	0.000E+00	0.000E+00	0.000E+00
PA233	4.661E-01	4.661E-01	4.662E-01	4.666E-01	4.670E-01	4.715E-01	4.770E-01	4.784E-01	4.786E-01	4.793E-01
PA234M	1.729E-01	1.626E-01	1.626E-01	1.626E-01	1.626E-01	1.625E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01
PA234	1.047E-02	3.177E-03	1.068E-03	2.118E-04	2.115E-04	2.112E-04	2.111E-04	2.111E-04	2.111E-04	2.111E-04
PA235	1.965E-07	1.995E-16	2.025E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U230	1.364E-07	1.342E-07	1.319E-07	1.187E-07	1.080E-07	5.020E-08	6.797E-09	3.387E-10	7.058E-13	0.000E+00
U231	4.113E-05	3.787E-05	3.487E-05	2.068E-05	1.296E-05	2.911E-07	1.458E-11	5.171E-18	2.738E-31	0.000E+00
U232	1.186E-02	1.188E-02	1.189E-02	1.200E-02	1.210E-02	1.288E-02	1.486E-02	1.768E-02	2.294E-02	4.694E-02
U233	2.939E-06	2.942E-06	2.945E-06	2.963E-06	2.978E-06	3.108E-06	3.449E-06	3.964E-06	5.025E-06	1.340E-05
U234	1.977E-02	1.980E-02	1.982E-02	1.997E-02	2.010E-02	2.119E-02	2.405E-02	2.836E-02	3.728E-02	1.071E-01
U235	4.921E-03	4.921E-03	4.921E-03	4.921E-03	4.921E-03	4.921E-03	4.922E-03	4.922E-03	4.922E-03	4.923E-03
U236	2.121E-01	2.121E-01	2.121E-01	2.121E-01	2.121E-01	2.121E-01	2.121E-01	2.121E-01	2.121E-01	2.122E-01
U237	1.113E+06	1.057E+06	1.004E+06	7.253E+05	5.422E+05	5.110E+04	1.098E+02	2.130E+00	2.069E+00	1.706E+00
U238	1.624E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01	1.624E-01
U239	1.564E+07	9.642E-03	5.946E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U240	2.615E+01	1.450E+01	8.038E+00	1.917E-01	6.775E-03	1.704E-06	1.704E-06	1.704E-06	1.704E-06	1.704E-06
U241	7.150E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NP235	1.127E-02	1.126E-02	1.125E-02	1.119E-02	1.113E-02	1.070E-02	9.629E-03	8.226E-03	5.948E-03	4.613E-04
NP236M	1.467E+01	1.013E+01	7.002E+00	6.736E-01	8.291E-02	3.415E-09	1.851E-28	0.000E+00	0.000E+00	0.000E+00
NP236	9.486E-06	9.486E-06	9.486E-06	9.486E-06	9.486E-06	9.486E-06	9.486E-06	9.486E-06	9.486E-06	9.486E-06
NP237	4.690E-01	4.694E-01	4.699E-01	4.723E-01	4.739E-01	4.781E-01	4.786E-01	4.786E-01	4.786E-01	4.793E-01
NP238	7.555E+05	6.414E+05	5.445E+05	1.931E+05	7.635E+04	4.101E+01	6.701E-02	6.693E-02	6.678E-02	6.557E-02
NP239	1.561E+07	1.357E+07	1.171E+07	4.613E+06	2.004E+06	2.347E+03	3.343E+01	3.343E+01	3.343E+01	3.342E+01
NP240M	5.331E+03	1.463E+01	8.109E+00	1.934E-01	6.835E-03	1.704E-06	1.704E-06	1.704E-06	1.704E-06	1.704E-06
NP240	2.590E+04	1.199E+01	5.551E-03	4.226E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NP241	7.150E-05	2.034E-18	6.845E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU236	1.310E+00	1.312E+00	1.313E+00	1.313E+00	1.310E+00	1.291E+00	1.240E+00	1.168E+00	1.032E+00	3.904E-01
PU237	7.525E+00	7.468E+00	7.411E+00	7.063E+00	6.765E+00	4.769E+00	1.916E+00	4.878E-01	2.920E-02	6.614E-12
PU238	6.050E+03	6.058E+03	6.064E+03	6.090E+03	6.100E+03	6.121E+03	6.154E+03	6.187E+03	6.215E+03	6.065E+03
PU239	2.508E+02	2.514E+02	2.519E+02	2.538E+02	2.545E+02	2.550E+02	2.550E+02	2.550E+02	2.550E+02	2.550E+02
PU240	5.383E+02	5.383E+02	5.383E+02	5.383E+02	5.383E+02	5.384E+02	5.385E+02	5.388E+02	5.392E+02	5.425E+02
PU241	8.848E+04	8.848E+04	8.847E+04	8.843E+04	8.840E+04	8.813E+04	8.744E+04	8.641E+04	8.432E+04	6.956E+04

PU242	1.815E+00	1.815E+00	1.815E+00	1.815E+00	1.815E+00	1.815E+00	1.815E+00	1.815E+00	1.815E+00	1.815E+00
PU243	4.917E+05	9.178E+04	1.713E+04	4.141E-01	3.342E-05	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06
PU244	1.706E-06	1.706E-06	1.706E-06	1.706E-06	1.706E-06	1.706E-06	1.706E-06	1.706E-06	1.706E-06	1.706E-06
PU245	3.805E+00	1.736E+00	7.922E-01	5.502E-03	6.446E-05	1.358E-20	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU246	1.484E-03	1.437E-03	1.392E-03	1.137E-03	9.488E-04	2.183E-04	4.725E-06	1.506E-08	1.087E-12	9.781E-13
AM239	1.229E-03	6.108E-04	3.036E-04	3.629E-06	6.913E-08	7.515E-22	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM240	3.564E-01	3.026E-01	2.569E-01	9.109E-02	3.602E-02	1.931E-05	5.671E-14	9.024E-27	0.000E+00	0.000E+00
AM241	1.002E+02	1.004E+02	1.006E+02	1.018E+02	1.029E+02	1.118E+02	1.349E+02	1.692E+02	2.385E+02	7.273E+02
AM242M	1.342E+01	1.342E+01	1.342E+01	1.342E+01	1.342E+01	1.341E+01	1.340E+01	1.339E+01	1.336E+01	1.311E+01
AM242	6.012E+04	3.578E+04	2.129E+04	8.073E+02	5.522E+01	1.334E+01	1.333E+01	1.332E+01	1.329E+01	1.305E+01
AM243	3.339E+01	3.342E+01	3.343E+01	3.343E+01	3.343E+01	3.343E+01	3.343E+01	3.343E+01	3.343E+01	3.342E+01
AM244M	2.875E+05	1.326E-03	6.112E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM244	1.508E+04	6.619E+03	2.905E+03	1.577E+01	1.483E-01	5.235E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM245	3.805E+00	2.141E+00	9.841E-01	6.838E-03	8.129E-05	1.120E-06	9.841E-07	8.099E-07	5.422E-07	2.290E-08

1 OUTPUT UNIT = 6
PAGE 233
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ ACTINIDES+DAUGHTERS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
AM246	1.484E-03	1.440E-03	1.394E-03	1.139E-03	9.504E-04	2.187E-04	4.732E-06	1.508E-08	1.087E-12	9.781E-13
CM241	2.340E-02	2.318E-02	2.296E-02	2.160E-02	2.045E-02	1.314E-02	4.137E-03	7.314E-04	2.066E-05	1.254E-17
CM242	4.066E+04	4.066E+04	4.062E+04	4.015E+04	3.967E+04	3.598E+04	2.789E+04	1.903E+04	8.671E+03	2.829E+01
CM243	3.462E+01	3.462E+01	3.462E+01	3.461E+01	3.461E+01	3.455E+01	3.441E+01	3.421E+01	3.379E+01	3.066E+01
CM244	9.388E+03	9.389E+03	9.388E+03	9.385E+03	9.383E+03	9.360E+03	9.301E+03	9.214E+03	9.037E+03	7.754E+03
CM245	1.206E+00	1.206E+00	1.206E+00	1.206E+00	1.206E+00	1.206E+00	1.206E+00	1.206E+00	1.206E+00	1.205E+00
CM246	4.814E-01	4.814E-01	4.814E-01	4.814E-01	4.814E-01	4.814E-01	4.814E-01	4.814E-01	4.814E-01	4.811E-01
CM247	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06	2.773E-06
CM248	1.448E-05	1.448E-05	1.448E-05	1.448E-05	1.448E-05	1.448E-05	1.448E-05	1.449E-05	1.449E-05	1.450E-05
CM249	7.502E-01	3.149E-04	1.354E-06	1.080E-06	9.676E-07	3.954E-07	3.828E-08	1.153E-09	8.537E-13	0.000E+00
CM250	3.681E-12	3.682E-12	3.683E-12	3.691E-12	3.699E-12	3.748E-12	3.830E-12	3.884E-12	3.909E-12	3.912E-12
CM251	2.177E-10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BK249	8.231E-02	8.233E-02	8.224E-02	8.168E-02	8.118E-02	7.725E-02	6.785E-02	5.584E-02	3.738E-02	1.579E-03
BK250	3.538E-01	2.678E-02	2.031E-03	4.657E-06	4.624E-06	4.364E-06	3.753E-06	2.993E-06	1.879E-06	4.771E-08
BK251	2.122E-04	3.344E-08	5.269E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CF249	2.954E-05	2.976E-05	2.999E-05	3.139E-05	3.264E-05	4.250E-05	6.602E-05	9.602E-05	1.420E-04	2.299E-04
CF250	1.329E-03	1.339E-03	1.339E-03	1.339E-03	1.338E-03	1.334E-03	1.322E-03	1.305E-03	1.270E-03	1.028E-03
CF251	1.030E-05	1.030E-05	1.030E-05	1.030E-05	1.030E-05	1.030E-05	1.030E-05	1.030E-05	1.029E-05	1.026E-05
CF252	3.451E-03	3.449E-03	3.448E-03	3.440E-03	3.433E-03	3.377E-03	3.234E-03	3.031E-03	2.653E-03	9.275E-04
CF253	4.088E-04	4.009E-04	3.932E-04	3.476E-04	3.113E-04	1.272E-04	1.232E-05	3.712E-07	2.747E-10	0.000E+00
CF254	7.880E-06	7.835E-06	7.791E-06	7.513E-06	7.274E-06	5.589E-06	2.810E-06	1.002E-06	1.200E-07	6.452E-15
CF255	1.325E-06	5.173E-09	2.021E-11	1.131E-26	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ES253	2.652E-04	2.675E-04	2.697E-04	2.797E-04	2.841E-04	2.351E-04	5.997E-05	4.275E-06	1.092E-08	4.176E-30
ES254M	2.973E-05	2.406E-05	1.947E-05	5.095E-06	1.536E-06	9.080E-11	8.472E-22	0.000E+00	0.000E+00	0.000E+00
ES254	4.704E-06	4.698E-06	4.692E-06	4.655E-06	4.622E-06	4.362E-06	3.751E-06	2.992E-06	1.878E-06	4.769E-08
ES255	9.949E-07	9.882E-07	9.794E-07	9.258E-07	8.804E-07	5.850E-07	2.014E-07	4.067E-08	1.512E-09	7.987E-21
SF250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TOTAL	3.415E+07	1.555E+07	1.345E+07	5.677E+06	2.767E+06	1.941E+05	1.320E+05	1.219E+05	1.094E+05	8.506E+04

0 CUMULATIVE TABLE TOTALS

AP+FP	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ACT+FP	3.415E+07	1.555E+07	1.345E+07	5.677E+06	2.767E+06	1.941E+05	1.320E+05	1.219E+05	1.094E+05	8.506E+04
AP+ACT+FP	3.415E+07	1.555E+07	1.345E+07	5.677E+06	2.767E+06	1.941E+05	1.320E+05	1.219E+05	1.094E+05	8.506E+04

1 OUTPUT UNIT = 6
PAGE 234
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
.0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
H 3	5.519E+02	5.519E+02	5.518E+02	5.515E+02	5.513E+02	5.494E+02	5.443E+02	5.368E+02	5.218E+02	4.168E+02
LI 6	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LI 7	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BE 9	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BE 10	2.966E-06	2.966E-06	2.966E-06	2.966E-06	2.966E-06	2.966E-06	2.966E-06	2.966E-06	2.966E-06	2.966E-06
C 14	1.196E-04	1.196E-04	1.196E-04	1.196E-04	1.196E-04	1.196E-04	1.196E-04	1.196E-04	1.196E-04	1.195E-04
NI 66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 66	3.736E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 67	8.318E-11	7.272E-11	6.357E-11	2.713E-11	1.266E-11	2.608E-14	2.564E-21	6.845E-32	0.000E+00	0.000E+00
ZN 67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 69	2.633E-02	1.042E-03	5.675E-04	1.234E-05	4.016E-07	3.377E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 69M	1.770E-03	9.669E-04	5.283E-04	1.149E-05	3.740E-07	3.145E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 69	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 70	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 70	3.176E-04	1.697E-14	9.070E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 70	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 71	3.749E-03	2.379E-08	2.850E-09	4.155E-15	2.492E-20	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 71M	3.931E-04	4.709E-05	5.642E-06	8.225E-12	4.934E-17	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 71	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 71	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 71M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 72	1.159E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI 72	2.166E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 72	4.223E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 72	4.988E+01	4.172E+01	3.488E+01	1.124E+01	4.077E+00	1.089E-03	5.184E-13	5.387E-27	0.000E+00	0.000E+00
GA 72	5.006E+01	4.793E+01	4.344E+01	1.597E+01	5.846E+00	1.562E-03	7.440E-13	7.731E-27	0.000E+00	0.000E+00
GE 72	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 73	3.813E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI 73	1.836E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 73	5.424E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 73	8.834E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 73	9.328E+01	1.699E+01	3.091E+00	6.345E-05	4.057E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 73	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 73M	9.336E+01	1.699E+01	3.091E+00	6.346E-05	4.058E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 74	7.192E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

NI 74	1.050E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 74	6.793E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 74	1.590E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 74	1.710E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 74	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 75	9.499E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI 75	4.227E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 75	6.560E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 75	2.806E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 75	3.422E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 75	3.475E+02	8.598E-01	2.070E-03	5.463E-20	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 75M	1.614E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 75	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI 76	1.058E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 235
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
CU 76	4.640E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 76	4.354E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 76	6.781E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 76	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 76	5.061E+01	3.690E+01	2.690E+01	3.635E+00	6.064E-01	2.946E-07	9.980E-24	0.000E+00	0.000E+00	0.000E+00
SE 76	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI 77	1.844E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 77	2.172E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 77	4.698E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 77	1.172E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 77	5.589E+02	2.679E+02	1.283E+02	1.213E+00	1.871E-02	3.690E-17	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 77M	1.191E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 77	1.507E+03	1.292E+03	1.079E+03	2.906E+02	8.640E+01	4.513E-03	3.047E-14	5.248E-31	0.000E+00	0.000E+00
SE 77	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 77M	5.659E+00	3.204E+00	2.675E+00	7.208E-01	2.143E-01	1.119E-05	7.556E-17	0.000E+00	0.000E+00	0.000E+00
NI 78	2.147E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 78	7.645E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 78	4.286E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 78	1.910E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 78	3.440E+03	1.111E+01	3.585E-02	5.976E-18	4.563E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 78	3.560E+03	8.338E+01	5.621E-01	8.905E-16	2.882E-29	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 78	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 79	2.573E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 79	3.297E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 79	2.330E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 79	7.140E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 79	8.292E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 79	4.112E-01	4.112E-01	4.112E-01	4.112E-01	4.112E-01	4.112E-01	4.112E-01	4.112E-01	4.112E-01	4.112E-01
SE 79M	8.339E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 79M	1.418E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 79	1.230E-06	9.693E-07	7.638E-07	1.688E-07	4.374E-08	7.579E-13	2.877E-25	0.000E+00	0.000E+00	0.000E+00
CU 80	2.642E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 80	1.113E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 80	2.065E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 80	1.272E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 80	1.737E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

SE 80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 80	6.848E-01	6.958E-02	1.060E-02	7.053E-08	1.647E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 80M	4.270E-01	6.502E-02	9.900E-03	6.591E-08	1.539E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CU 81	2.015E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 81	2.736E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 81	1.261E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 81	1.445E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 81	2.673E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 81	2.912E+04	1.816E-01	2.995E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 81M	7.452E+02	1.229E-01	2.028E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 81	7.645E-07	7.645E-07	7.645E-07	7.645E-07	7.645E-07	7.645E-07	7.645E-07	7.645E-07	7.645E-07	7.645E-07
KR 81M	3.544E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 236
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

+ FISSION PRODUCTS

POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
ZN 82	3.742E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 82	4.896E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 82	1.195E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 82	2.120E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 82M	9.228E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 82	7.151E+03	5.656E+03	4.469E+03	1.005E+03	2.645E+02	5.198E-03	2.740E-15	0.000E+00	0.000E+00	0.000E+00
BR 82M	2.790E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN 83	3.591E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 83	1.401E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 83	8.478E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 83	3.562E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 83	2.498E+04	5.846E-06	1.361E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 83M	3.539E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 83	6.215E+04	2.068E+03	6.370E+01	1.705E-08	4.643E-17	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 83	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 83M	6.275E+04	6.448E+03	2.466E+02	7.272E-08	1.981E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 84	2.104E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 84	3.265E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 84	2.987E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 84	9.739E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 84	1.020E+05	1.734E-02	2.649E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 84M	4.791E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 84	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GA 85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 85	7.362E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 85	1.635E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 85	5.330E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 85M	3.990E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 85	1.201E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 85	8.813E+03	8.813E+03	8.812E+03	8.807E+03	8.803E+03	8.767E+03	8.675E+03	8.537E+03	8.262E+03	6.379E+03
KR 85M	1.222E+05	1.932E+04	3.018E+03	2.363E-02	6.381E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 86	1.429E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 86	8.487E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 86	9.911E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

BR 86	8.274E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 86M	8.292E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 86	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 86	2.839E+03	2.787E+03	2.735E+03	2.432E+03	2.189E+03	9.314E+02	1.002E+02	3.539E+00	3.665E-03	0.000E+00
RB 86M	2.674E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 86	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GE 87	1.865E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 87	3.785E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 87	8.456E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 87	1.886E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 87	2.242E+05	3.270E+02	4.701E-01	4.802E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 87	2.088E-05	2.088E-05	2.088E-05	2.088E-05	2.088E-05	2.088E-05	2.088E-05	2.088E-05	2.088E-05	2.088E-05
SR 87	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 87M	1.374E+01	7.086E-01	3.655E-02	2.560E-10	1.294E-17	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 237
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
GE 88	9.634E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 88	4.320E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 88	3.235E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 88	1.888E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 88	3.142E+05	1.679E+04	8.967E+02	7.830E-06	4.822E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 88	3.220E+05	1.875E+04	1.001E+03	8.744E-06	5.384E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 88	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 89	4.885E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 89	9.797E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 89	1.222E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 89	3.696E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 89	4.056E+05	2.772E-09	1.526E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 89	4.241E+05	4.213E+05	4.184E+05	4.006E+05	3.853E+05	2.810E+05	1.233E+05	3.585E+04	2.820E+03	5.506E-06
Y 89	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 89M	3.884E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AS 90	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 90	2.774E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 90	7.360E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 90	3.633E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 90	3.860E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 90M	9.950E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	6.968E+04	6.968E+04	6.967E+04	6.966E+04	6.965E+04	6.954E+04	6.927E+04	6.886E+04	6.804E+04	6.186E+04
Y 90	7.418E+04	7.363E+04	7.315E+04	7.119E+04	7.039E+04	6.956E+04	6.929E+04	6.888E+04	6.806E+04	6.187E+04
Y 90M	1.394E+01	9.531E-01	6.514E-02	2.715E-09	6.769E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 90	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 90M	3.599E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 91	4.838E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 91	2.788E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 91	2.711E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 91	5.022E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 91	5.523E+05	2.305E+05	9.603E+04	3.751E+02	2.626E+00	8.470E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 91	5.741E+05	5.730E+05	5.705E+05	5.502E+05	5.320E+05	4.051E+05	1.990E+05	6.852E+04	7.633E+03	2.320E-04
Y 91M	3.207E+05	1.465E+05	6.103E+04	2.384E+02	1.669E+00	5.382E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 91	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 91	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 92	3.224E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 92	4.304E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

KR 92	1.440E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 92	4.496E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 92	6.229E+05	2.895E+04	1.345E+03	4.858E-06	1.358E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 92	6.269E+05	1.593E+05	1.982E+04	8.287E-03	1.364E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 92	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 92	3.154E-06	3.048E-06	2.946E-06	2.374E-06	1.956E-06	4.074E-07	6.796E-09	1.465E-11	4.754E-17	0.000E+00
SE 93	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 93	6.996E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 93	5.526E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 93	3.475E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 93	7.368E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 93	7.598E+05	3.376E+05	1.481E+05	8.042E+02	7.562E+00	2.669E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	1.771E+00	1.771E+00	1.771E+00	1.771E+00	1.771E+00	1.771E+00	1.771E+00	1.771E+00	1.771E+00	1.771E+00
NB 93	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 238
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
NB 93M	1.870E-01	1.871E-01	1.872E-01	1.879E-01	1.885E-01	1.933E-01	2.057E-01	2.241E-01	2.613E-01	5.235E-01
BR 94	6.213E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 94	1.704E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 94	1.878E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 94	7.052E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 94	7.869E+05	3.759E-06	1.688E-17	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 94	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 94	1.467E-04	1.467E-04	1.467E-04	1.467E-04	1.467E-04	1.467E-04	1.467E-04	1.467E-04	1.467E-04	1.466E-04
NB 94M	8.754E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 95	5.959E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 95	3.287E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 95	9.182E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 95	6.562E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 95	8.698E+05	2.048E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 95	9.019E+05	8.971E+05	8.922E+05	8.621E+05	8.361E+05	6.517E+05	3.402E+05	1.283E+05	1.725E+04	2.305E-03
NB 95	9.067E+05	9.066E+05	9.065E+05	9.047E+05	9.017E+05	8.403E+05	5.694E+05	2.533E+05	3.746E+04	5.118E-03
NB 95M	6.390E+03	6.381E+03	6.371E+03	6.261E+03	6.124E+03	4.834E+03	2.524E+03	9.519E+02	1.279E+02	1.710E-05
MO 95	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR 96	3.168E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 96	5.020E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 96	2.968E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 96	4.551E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 96	8.297E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 96	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 96	2.658E+03	1.862E+03	1.304E+03	1.366E+02	1.814E+01	1.388E-06	3.782E-25	0.000E+00	0.000E+00	0.000E+00
MO 96	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 97	3.277E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 97	5.823E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 97	2.437E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 97	7.227E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 97	9.525E+05	5.822E+05	3.559E+05	1.576E+04	9.691E+02	1.425E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 97	9.628E+05	5.861E+05	3.583E+05	1.584E+04	9.740E+02	1.536E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 97M	9.036E+05	5.515E+05	3.371E+05	1.493E+04	9.180E+02	1.350E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 97	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR 98	3.801E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB 98	1.425E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 98	9.912E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Y 98	5.117E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 98	9.644E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 98	9.846E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 98M	1.166E+04	7.210E-01	4.461E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC 98	8.723E-06	8.723E-06	8.723E-06	8.723E-06	8.723E-06	8.723E-06	8.723E-06	8.723E-06	8.723E-06	8.723E-06
RB 99	1.474E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 99	2.915E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y 99	2.951E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 99	9.558E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 99	1.001E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 99M	4.601E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 99	1.147E+06	1.012E+06	8.918E+05	4.015E+05	1.966E+05	5.969E+02	1.614E-04	2.270E-14	0.000E+00	0.000E+00
TC 99	1.199E+01	1.199E+01	1.200E+01	1.202E+01	1.202E+01	1.203E+01	1.203E+01	1.203E+01	1.203E+01	1.203E+01

1 OUTPUT UNIT = 6
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ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
TC 99M	1.005E+06	9.492E+05	8.528E+05	3.869E+05	1.895E+05	5.750E+02	1.555E-04	2.187E-14	0.000E+00	0.000E+00
RU 99	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB100	1.370E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR100	6.025E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y100	1.334E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR100	8.763E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB100	5.648E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB100M	5.648E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO100	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC100	5.795E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU100	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB101	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR101	8.695E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y101	4.397E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR101	5.568E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB101	9.434E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO101	1.029E+06	1.554E-09	2.326E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC101	1.029E+06	3.495E-08	7.051E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU101	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR102	8.344E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y102	1.148E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR102	3.233E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB102	8.203E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO102	1.007E+06	3.052E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC102	1.008E+06	3.076E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC102M	1.397E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU102	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH102	1.549E+00	1.548E+00	1.548E+00	1.545E+00	1.542E+00	1.519E+00	1.460E+00	1.377E+00	1.220E+00	4.688E-01
PD102	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR103	2.874E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y103	1.778E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR103	1.256E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB103	5.917E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO103	1.021E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC103	1.039E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU103	1.041E+06	1.032E+06	1.023E+06	9.672E+05	9.200E+05	6.131E+05	2.127E+05	4.346E+04	1.654E+03	1.053E-08
RH103	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

RH103M	9.381E+05	9.303E+05	9.222E+05	8.719E+05	8.294E+05	5.527E+05	1.917E+05	3.918E+04	1.491E+03	9.494E-09
SR104	1.236E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y104	1.705E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR104	3.171E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB104	2.960E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO104	8.516E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC104	9.150E+05	1.230E-06	1.518E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU104	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH104	9.749E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH104M	6.379E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD104	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y105	8.252E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR105	4.322E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB105	1.031E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6

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ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

+ FISSION PRODUCTS

POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
MO105	6.227E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC105	7.758E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU105	7.974E+05	1.265E+05	1.941E+04	1.362E-01	3.331E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH105	7.113E+05	6.376E+05	5.155E+05	1.168E+05	3.081E+04	6.159E-01	3.394E-13	1.369E-31	0.000E+00	0.000E+00
RH105M	2.233E+05	3.551E+04	5.452E+03	3.824E-02	9.353E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD105	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y106	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR106	4.374E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB106	2.652E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO106	3.416E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC106	5.617E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	4.814E+05	4.809E+05	4.805E+05	4.776E+05	4.751E+05	4.549E+05	4.064E+05	3.430E+05	2.420E+05	1.546E+04
RH106	5.407E+05	4.809E+05	4.805E+05	4.776E+05	4.751E+05	4.549E+05	4.064E+05	3.430E+05	2.420E+05	1.546E+04
RH106M	2.712E+04	6.184E+02	1.410E+01	5.624E-10	2.789E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD106	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG106	1.074E-06	1.031E-06	9.896E-07	7.644E-07	6.067E-07	9.298E-08	6.973E-10	4.529E-13	1.246E-19	0.000E+00
Y107	3.872E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR107	1.962E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB107	4.053E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO107	1.249E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC107	3.124E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU107	4.828E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH107	4.853E+05	6.278E-05	6.453E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD107	1.252E-01	1.252E-01	1.252E-01	1.252E-01	1.252E-01	1.252E-01	1.252E-01	1.252E-01	1.252E-01	1.252E-01
PD107M	1.508E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG107	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR108	7.060E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB108	9.151E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO108	3.578E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC108	1.829E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU108	3.351E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH108	3.384E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH108M	3.229E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD108	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG108	2.081E+00	3.262E-06	3.262E-06	3.261E-06	3.261E-06	3.260E-06	3.257E-06	3.253E-06	3.244E-06	3.174E-06
AG108M	3.665E-05	3.665E-05	3.665E-05	3.665E-05	3.664E-05	3.663E-05	3.660E-05	3.655E-05	3.645E-05	3.566E-05
CD108	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

ZR109	5.262E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB109	1.837E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO109	1.157E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC109	8.491E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU109	2.071E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH109	2.158E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH109M	1.079E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD109	3.137E+05	1.699E+05	9.157E+04	1.829E+03	5.514E+01	2.496E-11	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD109M	1.094E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG109	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG109M	3.136E+05	1.699E+05	9.160E+04	1.829E+03	5.517E+01	3.159E-03	2.888E-03	2.525E-03	1.915E-03	2.159E-04
CD109	3.304E-03	3.302E-03	3.299E-03	3.284E-03	3.270E-03	3.159E-03	2.888E-03	2.525E-03	1.915E-03	2.159E-04
NB110	2.083E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO110	2.296E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 241
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
TC110	2.088E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU110	8.992E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH110	9.696E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH110M	7.050E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD110	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG110	2.168E+05	8.416E+01	8.404E+01	8.331E+01	8.266E+01	7.755E+01	6.566E+01	5.115E+01	3.060E+01	5.317E-01
AG110M	6.337E+03	6.328E+03	6.319E+03	6.264E+03	6.215E+03	5.831E+03	4.937E+03	3.846E+03	2.301E+03	3.998E+01
CD110	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB111	1.773E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO111	4.727E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC111	6.715E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU111	3.899E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH111	5.322E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD111	5.542E+04	1.481E+02	3.264E+01	2.259E-03	4.287E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD111M	9.214E+02	2.032E+02	4.479E+01	3.101E-03	5.884E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG111	5.674E+04	5.430E+04	5.184E+04	3.861E+04	2.966E+04	3.490E+03	1.312E+01	3.031E-03	9.918E-11	0.000E+00
AG111M	5.588E+04	2.121E+02	4.676E+01	3.244E-03	6.156E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD111M	7.869E+01	2.789E-03	9.882E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB112	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO112	8.639E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC112	2.253E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU112	1.660E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH112	2.642E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD112	2.831E+04	1.872E+04	1.237E+04	9.000E+02	8.627E+01	4.664E-07	1.266E-28	0.000E+00	0.000E+00	0.000E+00
AG112	2.843E+04	2.181E+04	1.463E+04	1.066E+03	1.022E+02	5.524E-07	1.500E-28	0.000E+00	0.000E+00	0.000E+00
CD112	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO113	6.141E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC113	5.957E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU113	8.816E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH113	1.834E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD113	2.209E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG113	1.991E+04	4.165E+03	8.671E+02	4.182E-02	5.743E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG113M	2.240E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	7.447E+01	7.447E+01	7.447E+01	7.444E+01	7.441E+01	7.419E+01	7.361E+01	7.275E+01	7.102E+01	5.873E+01
IN113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

IN113M	3.401E-06	2.254E-08	1.494E-10	2.377E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO114	6.278E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC114	1.339E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU114	3.957E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH114	1.040E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD114	1.493E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG114	1.514E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD114	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	1.781E+01	6.436E+00	6.391E+00	6.114E+00	5.876E+00	4.258E+00	1.839E+00	5.215E-01	3.900E-02	5.112E-11
IN114M	6.772E+00	6.725E+00	6.678E+00	6.388E+00	6.140E+00	4.450E+00	1.921E+00	5.450E-01	4.075E-02	5.341E-11
SN114	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO115	4.205E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC115	2.827E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU115	1.869E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 242
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

FISSION PRODUCTS

+
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
RH115	7.914E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD115	1.425E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG115	1.068E+04	1.607E-07	2.339E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG115M	4.114E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115	1.475E+04	1.268E+04	1.085E+04	4.052E+03	1.678E+03	1.310E+00	1.025E-08	7.089E-21	0.000E+00	0.000E+00
CD115M	1.372E+03	1.361E+03	1.351E+03	1.286E+03	1.230E+03	8.605E+02	3.386E+02	8.360E+01	4.696E+00	6.449E-10
IN115	7.708E-12	7.736E-12	7.762E-12	7.865E-12	7.909E-12	8.021E-12	8.148E-12	8.210E-12	8.230E-12	8.231E-12
IN115M	1.477E+04	1.360E+04	1.177E+04	4.407E+03	1.825E+03	1.485E+00	2.380E-02	5.875E-03	3.300E-04	4.532E-14
SN115	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC116	1.753E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU116	4.179E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH116	3.756E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD116	1.060E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG116	5.910E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG116M	5.910E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD116	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN116	8.945E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN116M	6.516E+03	6.477E-01	6.438E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN116	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC117	7.416E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU117	6.058E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH117	1.863E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD117	9.223E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG117	5.796E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG117M	5.794E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD117	7.732E+03	3.171E+02	1.294E+01	2.054E-08	2.751E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD117M	4.183E+03	3.631E+02	3.144E+01	5.870E-06	5.598E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN117	7.129E+03	8.933E+02	6.378E+01	7.847E-06	7.435E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN117M	9.032E+03	1.208E+03	7.514E+01	6.090E-06	5.737E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN117	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN117M	1.709E+02	1.667E+02	1.626E+02	1.390E+02	1.208E+02	3.871E+01	1.987E+00	2.310E-02	2.408E-06	0.000E+00
TC118	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU118	4.131E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH118	3.217E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD118	7.902E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG118	7.615E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG118M	5.331E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

CD118	1.162E+04	5.714E-01	2.805E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN118	1.162E+04	5.724E-01	2.810E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN118M	5.083E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN118	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU119	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH119	2.592E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD119	5.851E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG119	1.076E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD119	5.809E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD119M	5.809E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN119	3.343E+03	8.337E-10	7.583E-22	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN119M	8.716E+03	1.436E-08	1.306E-20	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	2.195E+02	2.192E+02	2.189E+02	2.169E+02	2.152E+02	2.017E+02	1.702E+02	1.319E+02	7.811E+01	1.253E+00

1 OUTPUT UNIT = 6
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ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
RU120	8.636E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH120	3.914E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD120	2.194E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG120	7.384E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD120	1.141E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN120	5.823E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN120M	5.823E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN120	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH121	6.195E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD121	8.636E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG121	5.170E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD121	1.128E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN121	9.599E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN121M	2.378E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN121	1.202E+04	8.823E+03	6.469E+03	9.061E+02	1.561E+02	9.841E-05	6.581E-21	0.000E+00	0.000E+00	0.000E+00
SN121M	2.204E-01	2.204E-01	2.204E-01	2.203E-01	2.203E-01	2.201E-01	2.196E-01	2.189E-01	2.173E-01	2.056E-01
SB121	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH122	7.504E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD122	2.656E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG122	3.223E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD122	1.078E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN122	1.160E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN122M	8.238E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN122	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB122	3.266E+03	2.873E+03	2.527E+03	1.121E+03	5.416E+02	1.478E+00	3.026E-07	2.801E-17	0.000E+00	0.000E+00
SB122M	2.568E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE122	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH123	6.400E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD123	6.004E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG123	1.686E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD123	1.026E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN123	9.550E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN123M	4.007E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN123	2.616E+03	2.609E+03	2.602E+03	2.558E+03	2.520E+03	2.227E+03	1.614E+03	9.958E+02	3.685E+02	1.451E-01
SN123M	1.126E+04	4.447E-02	1.741E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB123	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE123	5.058E-12	5.063E-12	5.067E-12	5.096E-12	5.120E-12	5.307E-12	5.691E-12	6.067E-12	6.428E-12	6.616E-12

TE123M	4.755E+01	4.741E+01	4.727E+01	4.641E+01	4.566E+01	3.996E+01	2.823E+01	1.676E+01	5.733E+00	1.212E-03
PD124	1.121E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG124	7.684E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD124	9.313E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN124	1.518E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN124	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB124	2.157E+03	2.144E+03	2.132E+03	2.056E+03	1.990E+03	1.527E+03	7.650E+02	2.714E+02	3.215E+01	1.589E-06
SB124M	1.128E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE124	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG125	2.409E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD125	6.818E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN125	9.946E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN125M	7.266E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 244
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 .7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
SN125	9.861E+03	9.513E+03	9.177E+03	7.308E+03	5.961E+03	1.141E+03	1.526E+01	2.363E-02	3.879E-08	0.000E+00
SN125M	1.613E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	1.329E+04	1.328E+04	1.328E+04	1.327E+04	1.326E+04	1.310E+04	1.258E+04	1.183E+04	1.042E+04	3.829E+03
TE125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE125M	2.897E+03	2.898E+03	2.899E+03	2.905E+03	2.910E+03	2.939E+03	2.943E+03	2.843E+03	2.537E+03	9.342E+02
PD126	1.675E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG126	7.431E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD126	4.947E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN126	2.130E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01
SB126	1.127E+03	1.096E+03	1.066E+03	8.928E+02	7.620E+02	2.107E+02	7.464E+00	1.624E-01	1.145E-01	1.145E-01
SB126M	3.835E+02	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01	8.178E-01
TE126	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG127	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE126	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD127	2.813E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN127	1.265E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN127M	1.265E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN127	4.852E+04	9.244E+02	1.761E+01	2.245E-10	4.014E-20	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN127M	2.308E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB127	7.650E+04	7.094E+04	6.485E+04	3.667E+04	2.201E+04	3.500E+02	7.115E-03	6.520E-10	2.128E-24	0.000E+00
TE127	7.586E+04	7.426E+04	7.049E+04	4.490E+04	3.077E+04	8.800E+03	5.781E+03	3.262E+03	1.004E+03	9.272E-02
TE127M	1.007E+04	1.007E+04	1.006E+04	1.000E+04	9.894E+03	8.643E+03	5.902E+03	3.330E+03	1.025E+03	9.466E-02
I127	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE127	1.530E-01	1.516E-01	1.501E-01	1.414E-01	1.339E-01	8.644E-02	2.759E-02	4.973E-03	1.463E-04	1.221E-16
AG128	3.146E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD128	1.058E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN128	1.961E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN128	1.071E+05	2.271E+01	4.815E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB128	1.078E+04	4.281E+03	1.701E+03	4.927E+00	2.636E-02	9.563E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB128M	1.177E+05	2.757E+01	5.845E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE128	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I128	1.845E+04	3.895E-05	8.224E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE128	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD129	4.527E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN129	1.701E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN129	7.325E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

SN129M	7.763E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB129	2.136E+05	3.157E+04	4.602E+03	2.325E-02	4.239E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129	2.105E+05	5.785E+04	2.549E+04	1.871E+04	1.765E+04	1.098E+04	3.185E+03	4.975E+02	1.089E+01	8.861E-13
TE129M	3.118E+04	3.099E+04	3.069E+04	2.875E+04	2.712E+04	1.687E+04	4.893E+03	7.643E+02	1.673E+01	1.361E-12
I129	3.120E-02	3.121E-02	3.121E-02	3.122E-02	3.123E-02	3.129E-02	3.136E-02	3.139E-02	3.139E-02	3.139E-02
XE129	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE129M	1.796E+01	1.720E+01	1.647E+01	1.252E+01	9.795E+00	1.335E+00	7.374E-03	3.028E-06	3.239E-13	0.000E+00
CD130	3.708E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN130	1.663E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN130	2.206E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB130	7.005E+04	2.672E-01	1.019E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB130M	2.934E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE130	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I130	4.944E+04	2.533E+04	1.292E+04	1.822E+02	4.023E+00	1.451E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 245
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
I130M	1.956E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE130	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD131	6.167E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN131	6.134E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN131	1.836E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB131	4.972E+05	1.908E-04	7.195E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE131	5.382E+05	1.580E+04	1.198E+04	2.069E+03	4.299E+02	1.243E-03	4.415E-18	0.000E+00	0.000E+00	0.000E+00
TE131M	9.217E+04	7.019E+04	5.319E+04	9.189E+03	1.909E+03	5.520E-03	1.961E-17	0.000E+00	0.000E+00	0.000E+00
I131	6.086E+05	5.883E+05	5.661E+05	4.366E+05	3.430E+05	4.727E+04	2.680E+02	1.145E-01	1.328E-08	0.000E+00
XE131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M	6.851E+03	6.846E+03	6.833E+03	6.612E+03	6.259E+03	2.625E+03	1.066E+02	5.935E-01	1.226E-05	0.000E+00
CD132	5.638E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN132	1.525E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN132	9.402E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB132	2.842E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB132M	1.898E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE132	8.509E+05	7.654E+05	6.881E+05	3.508E+05	1.920E+05	1.439E+03	4.116E-03	1.991E-11	1.524E-28	0.000E+00
I132	8.680E+05	7.883E+05	7.090E+05	3.615E+05	1.978E+05	1.483E+03	4.241E-03	2.051E-11	1.570E-28	0.000E+00
XE132	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS132	2.953E+02	2.799E+02	2.653E+02	1.890E+02	1.396E+02	1.190E+01	1.928E-02	1.261E-06	3.072E-15	0.000E+00
BA132	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN133	1.894E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN133	2.853E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB133	3.108E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE133	6.957E+05	8.175E+00	1.000E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE133M	3.981E+05	4.875E+01	5.966E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I133	1.181E+06	8.091E+05	5.424E+05	4.309E+04	4.470E+03	4.585E-05	6.620E-26	0.000E+00	0.000E+00	0.000E+00
I133M	4.125E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	1.139E+06	1.130E+06	1.102E+06	7.934E+05	5.536E+05	2.676E+04	9.643E+00	6.594E-05	1.541E-15	0.000E+00
XE133M	3.802E+04	3.663E+04	3.407E+04	1.545E+04	6.548E+03	4.570E+00	2.581E-08	1.096E-20	0.000E+00	0.000E+00
CS133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN134	1.154E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN134	4.554E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB134	5.772E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB134M	5.317E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE134	8.922E+05	5.828E+00	3.804E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

I134	1.280E+06	3.370E+02	2.709E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I134M	1.452E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE134	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE134M	1.057E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	2.578E+05	2.577E+05	2.576E+05	2.569E+05	2.562E+05	2.508E+05	2.374E+05	2.185E+05	1.842E+05	4.802E+04
CS134M	6.292E+04	3.574E+03	2.030E+02	2.621E-06	2.290E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA134	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN135	5.139E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB135	2.930E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE135	4.832E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I135	1.106E+06	3.145E+05	8.938E+04	3.095E+01	2.480E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE135	2.341E+05	4.391E+05	2.738E+05	1.479E+03	8.668E+00	4.557E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE135M	2.454E+05	5.038E+04	1.432E+04	4.958E+00	3.972E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS135	4.466E-01	4.468E-01	4.469E-01	4.471E-01	4.471E-01	4.471E-01	4.471E-01	4.471E-01	4.471E-01	4.471E-01

1 OUTPUT UNIT = 6
PAGE 246
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

+ FISSION PRODUCTS

POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
CS135M	6.179E+04	5.028E+00	4.092E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA135	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA135M	3.360E+02	2.514E+02	1.882E+02	3.001E+01	5.805E+00	9.394E-06	7.345E-21	0.000E+00	0.000E+00	0.000E+00
SN136	4.767E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB136	6.170E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE136	2.449E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I136	5.174E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I136M	2.984E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS136	7.090E+04	6.905E+04	6.725E+04	5.687E+04	4.896E+04	1.450E+04	6.065E+02	5.188E+00	2.875E-04	0.000E+00
BA136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA136M	1.168E+04	1.138E+04	1.108E+04	9.373E+03	8.068E+03	2.390E+03	9.994E+01	8.550E-01	4.738E-05	0.000E+00
SB137	1.028E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE137	7.399E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I137	4.868E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE137	1.024E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	1.045E+05	1.045E+05	1.045E+05	1.045E+05	1.044E+05	1.043E+05	1.039E+05	1.033E+05	1.021E+05	9.308E+04
BA137	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	9.900E+04	9.884E+04	9.884E+04	9.882E+04	9.880E+04	9.865E+04	9.828E+04	9.772E+04	9.658E+04	8.806E+04
SB138	1.343E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE138	1.910E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I138	2.378E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE138	9.230E+05	4.680E-10	2.368E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS138	1.038E+06	3.286E-01	6.102E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS138M	5.405E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA138	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA138	8.045E-11	8.045E-11	8.045E-11	8.045E-11	8.045E-11	8.045E-11	8.045E-11	8.045E-11	8.045E-11	8.045E-11
SB139	1.008E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE139	3.892E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I139	1.050E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE139	7.074E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS139	9.813E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA139	1.023E+06	2.770E+03	6.629E+00	1.671E-16	2.282E-31	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE140	4.956E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

I140	2.874E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE140	4.450E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS140	8.812E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA140	9.865E+05	9.602E+05	9.345E+05	7.872E+05	6.751E+05	1.941E+05	7.512E+03	5.721E+01	2.496E-03	0.000E+00
LA140	1.092E+06	1.070E+06	1.047E+06	8.981E+05	7.745E+05	2.234E+05	8.645E+03	6.584E+01	2.872E-03	0.000E+00
CE140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR140	2.934E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE141	2.307E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I141	4.977E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE141	1.587E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS141	6.454E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA141	9.252E+05	1.282E-06	1.748E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA141	9.303E+05	1.216E+05	1.465E+04	2.214E-02	1.372E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	9.347E+05	9.293E+05	9.199E+05	8.599E+05	8.095E+05	4.958E+05	1.380E+05	2.025E+04	3.901E+02	1.158E-11

1 OUTPUT UNIT = 6
PAGE 247
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

FISSION PRODUCTS

POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
PR141	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND141	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE142	2.642E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I142	8.280E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE142	5.551E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS142	3.797E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA142	8.632E+05	4.791E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA142	8.865E+05	4.589E+03	2.105E+01	3.276E-14	1.845E-27	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	2.732E-05	2.732E-05	2.732E-05	2.732E-05	2.732E-05	2.732E-05	2.732E-05	2.732E-05	2.732E-05	2.732E-05
PR142	1.085E+05	7.039E+04	4.557E+04	2.903E+03	2.471E+02	5.091E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR142M	2.125E+04	3.034E-11	4.333E-26	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I143	5.727E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE143	9.947E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS143	1.776E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA143	7.401E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA143	8.262E+05	2.768E-10	9.132E-26	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE143	8.346E+05	6.533E+05	5.078E+05	1.029E+05	2.466E+04	2.273E-01	1.663E-14	0.000E+00	0.000E+00	0.000E+00
PR143	8.061E+05	8.045E+05	7.988E+05	7.165E+05	6.272E+05	1.945E+05	9.066E+03	9.123E+01	7.064E-03	0.000E+00
ND143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I144	4.844E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE144	1.861E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS144	5.749E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA144	5.410E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA144	7.182E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE144	7.621E+05	7.611E+05	7.602E+05	7.544E+05	7.492E+05	7.083E+05	6.119E+05	4.913E+05	3.127E+05	8.871E+03
PR144	7.698E+05	7.612E+05	7.602E+05	7.544E+05	7.492E+05	7.083E+05	6.119E+05	4.913E+05	3.128E+05	8.872E+03
PR144M	9.155E+03	9.134E+03	9.123E+03	9.052E+03	8.990E+03	8.500E+03	7.343E+03	5.896E+03	3.753E+03	1.065E+02
ND144	1.477E-09	1.478E-09	1.478E-09	1.480E-09	1.482E-09	1.497E-09	1.533E-09	1.578E-09	1.644E-09	1.757E-09
I145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE145	2.297E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS145	1.481E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA145	2.883E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA145	5.313E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE145	5.751E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR145	5.754E+05	1.446E+05	3.599E+04	5.379E+00	2.032E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

PM145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE146	1.455E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS146	2.276E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA146	1.097E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA146	3.528E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE146	4.688E+05	2.576E-10	1.404E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR146	4.709E+05	1.263E-03	1.397E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND146	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM146	3.036E+00	3.035E+00	3.035E+00	3.032E+00	3.029E+00	3.005E+00	2.943E+00	2.853E+00	2.676E+00	1.617E+00
SM146	3.069E-07	3.069E-07	3.070E-07	3.071E-07	3.071E-07	3.078E-07	3.096E-07	3.122E-07	3.174E-07	3.482E-07
XE147	1.331E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS147	4.088E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA147	2.652E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 248
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
LA147	1.749E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE147	3.666E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR147	3.773E+05	3.640E-13	3.194E-31	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND147	3.839E+05	3.724E+05	3.609E+05	2.959E+05	2.478E+05	5.862E+04	1.365E+03	4.848E+00	4.400E-05	0.000E+00
PM147	6.156E+04	6.167E+04	6.178E+04	6.239E+04	6.282E+04	6.394E+04	6.187E+04	5.798E+04	5.071E+04	1.762E+04
SM147	1.016E-06	1.016E-06	1.017E-06	1.021E-06	1.024E-06	1.050E-06	1.117E-06	1.212E-06	1.391E-06	2.202E-06
CS148	2.736E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA148	5.043E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA148	6.785E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE148	2.741E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR148	3.064E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND148	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.689E+05	1.584E+05	1.486E+05	9.897E+04	6.887E+04	4.035E+03	1.981E+02	4.335E+01	1.935E+00	4.331E-11
PM148M	1.579E+04	1.566E+04	1.553E+04	1.472E+04	1.404E+04	9.544E+03	3.486E+03	7.696E+02	3.435E+01	7.690E-10
SM148	6.625E-11	6.627E-11	6.629E-11	6.639E-11	6.646E-11	6.664E-11	6.673E-11	6.677E-11	6.678E-11	6.678E-11
CS149	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA149	6.095E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA149	1.775E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE149	1.558E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR149	2.207E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND149	2.441E+05	2.034E+03	1.661E+01	9.907E-13	1.457E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM149	4.154E+05	3.622E+05	3.098E+05	1.148E+05	4.725E+04	3.500E+01	2.388E-07	1.345E-19	0.000E+00	0.000E+00
SM149	3.155E-13	3.526E-13	3.845E-13	5.026E-13	5.435E-13	5.721E-13	5.721E-13	5.721E-13	5.721E-13	5.721E-13
EU149	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS150	2.631E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA150	4.687E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA150	3.571E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE150	7.545E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR150	1.543E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND150	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM150	6.587E+03	2.957E+02	1.327E+01	3.857E-08	8.875E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM150	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU150	1.969E-05	1.969E-05	1.969E-05	1.969E-05	1.968E-05	1.966E-05	1.960E-05	1.951E-05	1.932E-05	1.788E-05
BA151	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA151	4.723E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE151	2.320E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR151	8.761E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

ND151	1.349E+05	4.492E-13	1.483E-30	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM151	1.348E+05	1.013E+05	7.557E+04	1.181E+04	2.246E+03	3.148E-03	1.693E-18	0.000E+00	0.000E+00	0.000E+00
SM151	3.303E+02	3.316E+02	3.325E+02	3.348E+02	3.351E+02	3.350E+02	3.346E+02	3.340E+02	3.327E+02	3.226E+02
EU151	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA152	1.287E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA152	5.530E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE152	5.244E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR152	3.944E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND152	9.263E+04	1.326E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM152	9.544E+04	2.060E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM152M	1.853E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM152	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU152	4.474E+00	4.474E+00	4.474E+00	4.472E+00	4.470E+00	4.456E+00	4.419E+00	4.364E+00	4.252E+00	3.468E+00
EU152M	8.888E+01	3.641E+01	1.491E+01	5.227E-02	3.325E-04	4.914E-22	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1
PAGE 249 OUTPUT UNIT = 6
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
GD152	9.599E-13	9.603E-13	9.605E-13	9.607E-13	9.607E-13	9.612E-13	9.625E-13	9.645E-13	9.684E-13	9.959E-13
LA153	5.886E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE153	9.672E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR153	1.284E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND153	5.516E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM153	6.292E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM153	4.937E+05	4.133E+05	3.458E+05	1.119E+05	4.079E+04	1.127E+01	5.868E-09	6.973E-23	0.000E+00	0.000E+00
EU153	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153	2.127E+01	2.123E+01	2.120E+01	2.101E+01	2.084E+01	1.951E+01	1.643E+01	1.270E+01	7.471E+00	1.138E-01
LA154	2.739E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE154	1.150E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR154	3.197E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND154	2.996E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM154	3.704E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM154M	6.436E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM154	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU154	1.366E+04	1.366E+04	1.366E+04	1.365E+04	1.364E+04	1.357E+04	1.339E+04	1.313E+04	1.260E+04	9.128E+03
GD154	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA155	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE155	1.344E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR155	6.562E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND155	1.183E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM155	2.400E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM155	3.037E+04	5.398E-06	9.313E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU155	9.289E+03	9.287E+03	9.285E+03	9.274E+03	9.264E+03	9.183E+03	8.975E+03	8.671E+03	8.077E+03	4.618E+03
GD155M	1.678E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD155	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE156	1.277E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR156	1.262E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND156	4.411E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM156	1.380E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM156	1.846E+04	7.626E+03	3.148E+03	1.158E+01	7.695E-02	1.617E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU156	3.088E+05	3.021E+05	2.954E+05	2.557E+05	2.247E+05	7.863E+04	5.084E+03	8.356E+01	1.777E-02	0.000E+00
GD156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE157	9.925E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR157	1.931E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND157	1.257E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

PM157	6.784E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM157	1.233E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU157	3.309E+04	1.921E+04	1.111E+04	3.473E+02	1.563E+01	1.828E-10	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD157	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR158	1.411E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND158	1.918E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM158	2.224E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM158	6.757E+03	7.983E-02	9.427E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU158	7.275E+03	1.248E+00	3.679E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD158	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR159	5.490E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND159	1.886E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM159	5.110E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM159	3.225E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

1 OUTPUT UNIT = 6
PAGE 250
ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities
+ FISSION PRODUCTS
POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC
0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES
One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
EU159	3.909E+03	4.746E-09	5.031E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD159	7.207E+03	4.655E+03	2.976E+03	1.753E+02	1.390E+01	1.619E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB159	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND160	1.372E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM160	8.252E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM160	1.168E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU160	1.791E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD160	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB160	1.746E+03	1.738E+03	1.729E+03	1.678E+03	1.633E+03	1.310E+03	7.368E+02	3.109E+02	5.265E+01	4.351E-05
DY160	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND161	9.821E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM161	9.771E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM161	3.211E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU161	7.800E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD161	9.253E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB161	1.412E+03	1.344E+03	1.278E+03	9.306E+02	7.006E+02	6.998E+01	1.716E-01	2.087E-05	1.824E-13	0.000E+00
DY161	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM162	4.178E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM162	4.064E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU162	2.265E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD162	4.149E+02	1.276E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB162	4.109E+02	4.942E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB162M	1.261E+01	3.227E-01	7.743E-03	4.262E-13	2.820E-22	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
DY162	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM163	4.508E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU163	5.245E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD163	1.656E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB163	1.789E+02	1.488E-09	1.142E-20	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB163M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
DY163	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM164	3.880E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU164	1.042E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD164	6.515E+01	6.580E-09	6.645E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB164	8.093E+01	7.637E-09	7.712E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
DY164	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM165	2.392E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU165	1.617E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

GD165	2.172E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB165	3.550E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
DY165	1.006E+03	2.937E+01	8.526E-01	1.568E-10	3.052E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
DY165M	6.297E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
HO165	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
DY166	2.805E+01	2.533E+01	2.287E+01	1.198E+01	6.720E+00	6.146E-02	2.950E-07	3.101E-15	1.141E-31	0.000E+00
HO166	3.211E+02	2.425E+02	1.842E+02	3.888E+01	1.363E+01	9.157E-02	4.395E-07	4.621E-15	1.597E-31	0.000E+00
HO166M	8.586E-03	8.586E-03	8.586E-03	8.586E-03	8.586E-03	8.585E-03	8.585E-03	8.583E-03	8.581E-03	8.561E-03
ER166	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER167	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER167M	9.993E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER168	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
YB168	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER169	5.107E-01	4.922E-01	4.744E-01	3.756E-01	3.048E-01	5.590E-02	6.698E-04	8.792E-07	1.027E-12	0.000E+00

1 OUTPUT UNIT = 6

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ORIGEN2 V2.1 (8-1-91), Run on 11/30/01 at 16:51:51

* AP1000 UO2 Case - Decayed Average Assembly Activities

+ FISSION PRODUCTS

POWER= 2.16543E+01 MW, BURNUP= 3.38110E+04 MWD, FLUX= 3.82E+14 N/CM**2-SEC

0 7 NUCLIDE TABLE: RADIOACTIVITY, CURIES

One Asy at 4.728 w/o; Region-wise Power to 54 GWD/MTU

	2xBurned	12.0HR	1.0D	100.0HR	7.0D	30.0D	90.0D	180.0D	1.0YR	5.0YR
TM169	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
YB169	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER170	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TM170	1.449E-01	1.445E-01	1.441E-01	1.417E-01	1.395E-01	1.232E-01	8.919E-02	5.491E-02	2.023E-02	7.686E-06
TM170M	1.629E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
YB170	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER171	2.236E-06	7.398E-07	2.448E-07	2.219E-10	4.198E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TM171	4.107E-03	4.105E-03	4.103E-03	4.090E-03	4.079E-03	3.987E-03	3.758E-03	3.438E-03	2.863E-03	6.755E-04
YB171	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ER172	9.132E-10	7.706E-10	6.503E-10	2.219E-10	8.481E-11	3.446E-14	4.905E-23	0.000E+00	0.000E+00	0.000E+00
TM172	5.744E-04	5.040E-04	4.422E-04	1.931E-04	9.205E-05	2.249E-07	3.437E-14	2.054E-24	0.000E+00	0.000E+00
YB172	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TOTAL	1.033E+08	2.587E+07	2.221E+07	1.528E+07	1.308E+07	7.809E+06	4.573E+06	2.940E+06	1.800E+06	4.450E+05

0 CUMULATIVE TABLE TOTALS

AP+FP	1.033E+08	2.587E+07	2.221E+07	1.528E+07	1.308E+07	7.809E+06	4.573E+06	2.940E+06	1.800E+06	4.450E+05
ACT+FP	1.374E+08	4.142E+07	3.566E+07	2.096E+07	1.585E+07	8.003E+06	4.705E+06	3.062E+06	1.909E+06	5.301E+05
AP+ACT+FP	1.374E+08	4.142E+07	3.566E+07	2.096E+07	1.585E+07	8.003E+06	4.705E+06	3.062E+06	1.909E+06	5.301E+05

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(Sprung et al. 2000)

NUREG/CR-6672, Vol. 1
SAND2000-0234

Reexamination of Spent Fuel Shipment Risk Estimates

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NUREG/CR-6672, Vol. 1
SAND2000-0234

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Manuscript Completed: February 2000
Date Published: March 2000

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Prepared for
Spent Fuel Project Office
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code J5160