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EVALUATION OF FSV-1 CASK FOR THE TRANSPORT OF LWR IRRADIATED FUEL ASSEMBLIES

by
PROJECT STAFF

DATE PUBLISHED: MAY 1980

GENERAL ATOMIC COMPANY

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ABSTRACT

The Model FSV-1 spent fuel shipping cask was designed by General Atomic Company (GA) to service the Fort St. Vrain (FSV) nuclear generating station, a High Temperature Gas Reactor (HTGR) owned and operated by Public Service Company of Colorado (PSC). This report presents an evaluation of the suitability of the FSV-1 cask for the transport of irradiated Light Water Reactor (LWR) fuel assemblies from both Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR). The FSV-1 cask evaluation parameters covered a wide spectrum of LWR fuel assemblies, based on burnup in Megawatt Days/Metric Ton of Heavy Metal (MWD/MTHM) and years of decay since irradiation. The criteria for suitability included allowable radiation dose rates, cask surface and interior temperatures and the Gross Vehicle Weight (GVW) of the complete shipping system.

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Summary

Based on this evaluation, the FSV-1 cask can be modified to be suitable for the transport of a significant spectrum of LWR irradiated fuel assemblies (Figure 1).

During the evaluation it was recognized that obtaining a revised Certificate of Compliance that added LWR fuel assemblies to the authorized contents of the FSV-1 cask would constitute final approval of its suitability. Therefore, those regulatory requirements of 10CFR71 and 49CFR173, concerning containment and radiation would control the design modifications to the cask and the selection of suitable LWR irradiated fuel assemblies based on burnup and decay. Another concern outside of the Code of Federal Regulations was the impact of the modifications and the contents on the Gross Vehicle Weight (GVW) and any subsequent decrease in schedule flexibility when conducting overweight shipments. One further and unique constraint on any modifications to FSV-1 cask unit number 3, is a contractual obligation with Public Service Company of Colorado (PSC) that this cask be made available, in case of spent fuel shipping problems, in the configuration for shipping HTGR fuel elements.

In addition to these required criteria, a desirable modification was the removal of the inner container to make available the largest possible cavity (17 3/4 inches in diameter by 188 inches long). Thus, the evaluation consisted of trade-offs between mandatory requirements, operational concerns, contractual constraints, and desirable features.

The containment requirement may be satisfied by the canister containing the PWR or BWR fuel assemblies, or by the modifed FSV-1 cask closures or by some combination of both. The final identification of the containment boundaries will be determined when the design of the canisters is more definitive.

Allowable radiation levels for the loaded cask are found in 49CFR173.393, thus the gamma and neutron shielding capability of the cask will determine the selection of fuel assemblies that can be transported. Since the addition of shielding adds weight, the consideration of an acceptable GVW really controls the spectrum of acceptable LWR fuels. For this evaluation an upper limit of 78,000 pounds GVW was selected. At this weight, shipping operations without permits are possible in all except 10 states and the District of Columbia.

SPECTRUM OF LWR IRRADIATED FUEL ASSEMBLIES (F/A)
THAT CAN BE TRANSPORTED IN THE MODIFIED FSV-1
CASK - BASED ON 10 MREM/HR AT 6 FEET FROM TRANSPORT
VEHICLE.
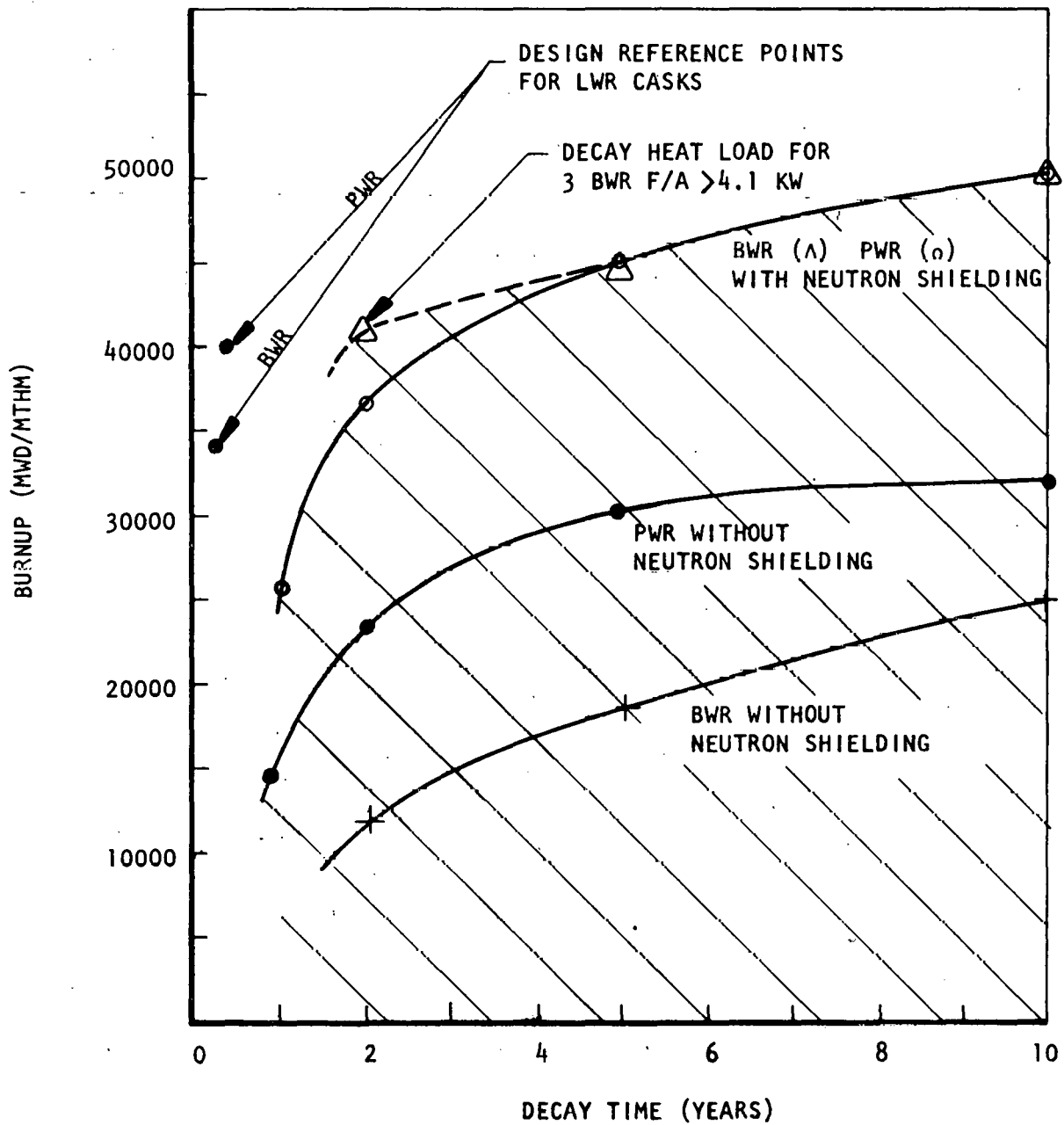
Figure 1

Although many states allow 80,000 pound GVW shipments without permits, the maximum axle loadings and "bridge" formulas make it difficult to operate a 5 axle tractor and semi-trailer combination in excess of 78,000 pounds, GVW.

Typical casks designed for the transport of LWR fuel assemblies can accept fuel heat loads of 10 to 12 KW compared to 4.1 KW for the FSV-1 cask. This limit of 4.1 KW, was based on the expected heat load from 6 fuel elements from the HTGR reactor rather than the capacity of cask and contents. As the evaluation developed it was apparent that the spectrum of LWR irradiated fuel elements that could be transported would be limited by shielding considerations rather than the 4.1 KW fuel heat load limit except for the case of 3 BWR fuel assemblies with 42,000 MWD/MTHM burnup and 2 years decay time, which have a heat load of 4.2 KW. This is a very small area of the plot (Figure 1) and with slightly less burnup or slightly more decay this fuel heat load will be less than 4.1 KW. No potential problems with the temperature of the canister, the inner closure seal assembly or the Boro-Silicone neutron shielding were detected during the thermal studies. Accessible exterior surface temperatures of casks during transport are limited to 180°F by 49CFR173. Certain combinations of solar heat load, fuel heat load, and high ambient temperatures (generally in excess of 100°F) will result in exterior surface temperatures in excess of 180°F, therefore, a personnel barrier will be included in the modified design to make the exterior surfaces of the FSV-1 cask inaccessible. It may be desirable to make this personnel barrier removable and thus only use it when required, and thus simplify handling operations.

This evaluation did not include a detailed structural analysis because of the constraints of schedule, budget and the preliminary nature of information concerning the canisters, however, careful attention was given the modified design in regard to maintaining the existing load paths and minimizing the increase in the total package (cask and contents) weight. Although additional structural analysis will be required to support the request for a revised Certificate of Compliance, this evaluation to date has not produced any serious concerns.

LWR fuel assemblies when compared to HTGR fuel elements have differences in quantities of heavy metal, number of curies of fission products and neutron sources as listed below:

The FSV cask, during the transport of 6 HTGR irradiated fuel elements contains as a design basis:

$$7.8 \text{ Kg of } U_{235}$$

$$67.8 \text{ Kg of Thorium}$$

$$1 \times 10^6 \text{ Curies of fission products}$$

A typical LWR cask during the transport of 1 PWR or 2 BWR irradiated fuel assemblies contains as a design basis:

$$4.0 \text{ Kg of } U_{235} \text{ and Plutonium}$$

$$3 \times 10^6 \text{ Curies of fission products}$$

$$9 \times 10^9 \text{ n/sec neutron source}$$

These differences do not appear to have any impact on the obtaining of a revised Certificate of Compliance, based on this evaluation.

No design features, material properties, or existing regulations were identified that would preclude the issuance of a revision to the Certificate of Compliance to add a significant spectrum of LWR irradiated fuel assemblies to the authorized contents for a modified FSV-1 cask.

1.0 TASK DESCRIPTION

1.1 Scope

Evaluate the General Atomic Company, model FSV-1 Spent Fuel Shipping Cask to assess its suitability for the transport of Light Water Reactor (LWR) irradiated fuel assemblies. Identify any required modifications and assess the acceptability of a request to amend the Certificate of Compliance to add LWR irradiated fuel assemblies to the authorized contents.

Phase I Evaluation of LWR Fuel Shipping

- Calculate detailed neutron-gamma shielding for PWR/BWR fuel with age (1 to 10 years old) as parameter.
- Use scoping thermal calculations to assess spent fuel package wall temperatures and fuel basket design requirements.
- Assess containment adequacy (alternate head arrangement, seal adequacy, leak-check capability, etc.).
- Analyze cask and vehicle weight in detail.
- Assess current status of state restrictions regarding 73,280 lb vs. 80,000 lb GVW.
- Prepare designlayouts and sketches to support technical assessment of feasibility and cost estimate.
- Perform structural engineering to insure package limitations are not exceeded.
- Prepare a detailed program cost and schedule proposal for design certificate amendment, and acquisition of the FSV cask, with supporting technical documentation.

Note: Phase II - Evaluation of HLW Shipping will be the subject of a later report.

1.2 Assumptions

This evaluation of the FSV-1 cask was based on the following information and/or assumptions:

- Neutron and gamma source terms supplied by Sandia Laboratories utilizing the SANDIA-ORIGEN computer code.

- The inner container of the FSV-1 cask is removed.
- LWR irradiated fuel assemblies will be in welded, gas tight canisters.
- The closure system will be modified to provide an inner closure which does not require an inner container.
- Criticality control can be accomplished by administrative methods or poison inserts. A criticality study will not be a part of this evaluation.
- The following ground scatter corrections factors were used:
  For neutron dose rate - 1.4
  For gamma dose rate - 1.1
- The reference PWR fuel assembly canister was assumed to be 12.75 inches O.D. by 169.89 inches long, without the pintle.
- The reference BWR fuel assembly canister was assumed to be 16.00 inches O.D. by 169.89 inches long without the pintle.
- The maximum allowable dose rate will be 10 mrem/hr at a distance of 6 feet from the edge of the vehicle (49CFR173.393).
- The maximum allowable dose rate on the surface of the inner closure will be 50 mrem/hr.

1.3 Discussion

In the data supplied, the gamma source term for Ag 110m, assumed the emission of a 2.75 MeV photon 100% of the time. This assumption results in an over estimation of the gamma dose rate. The amount of this overestimation varies with decay time and with the thickness of the gamma and neutron shielding. Also, in comparison with other values currently used in the nuclear industry, the SANDIA-ORIGEN code, (α·n) conversion is believed to be high by a factor of 2.

Unfortunately these concerns about the supplied data were not dis- covered until the shielding study was nearly complete. Schedule and budget constraints did not allow the shielding study to be redone, however, the appropriate corrections were incorporated into the results presented in the shielding evaluation (Section 2.3 and in the Summary (figure 1).

The information used to define the size, shape, and weight of the PWR and BWR irradiated fuel assembly canisters, especially in the case of BWR fuel, was obtained from preliminary design concepts, therefore these envelopes should be verified and/or updated as soon as better information becomes available.

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2.0 LICENSING EVALUATION

Prior to any use of the FSV-1 cask for the transport of LWR irradiated fuel assemblies, the Certificate of Compliance (COC) must be amended to add the LWR fuel assemblies to the authorized contents. During this evaluation, particular attention was directed toward making those hardware changes and selecting the LWR fuel assemblies that have the least impact on the existing COC and thus there is a high probability that a revised COC will be obtained. The following subsections summarize the structural, thermal, shielding, and containment evaluation.

2.1 Structural Evaluation

The model FSV-1 cask consists of an inner container which provides for the containment of the radioactive material and an outer body which provides the gamma shielding and the structural integrity. The current operational configuration is shown on drawing GADR 55-2-2 (Ref. 7.2).

When the FSV-1 cask is used to ship canisterized LWR fuel assemblies, the inner container will be removed to provide a larger cavity, and since the inner closure is part of the inner container, the cask body will be modified to accommodate an inner closure that provides the gamma shielding and an outer closure that provides the structural integrity. The proposed modification to the closure system is shown on drawing SK3325-100, Issue B, Sheet 5 (Ref. 6).

The new inner closure has an attached seal assembly consisting of concentric silicone rubber seals. An external port on the upper surface of this closure allows for the evacuation of the seal interspace and thus verification that the closure is sealed. The inner closure further has provisions for sampling the contents of the cask cavity.

An impact limiter, similar to the existing one but slightly modified to accommodate the new outer closure configuration, provides protection for the closure system during the accident conditions of transport. These above described modifications will make the FSV-1 cask suitable for the transportation of some canisterized LWR irradiated fuel assemblies, based

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on burnup and decay. To extend the spectrum of LWR fuel that can be
transported i.e.: higher burnup and shorter decay times, neutron shielding
must be added to the exterior of the cask. Several different neutron
shielding materials were considered in various thicknesses. Based on the
feasibility of fabrication, the acceptable weight increase and a reasonable
increase in the spectrum of LWR fuel assemblies which could be transported,
a 3 inch layer of Boro-Silicone, protected with a stainless steel shell
was selected for the conceptual design.

The modified closure design as evaluated was devised to minimize
the amount of new structural analysis required to support the request
for a revision to the Certificate of Compliance. All load paths have
remained the same or equivalent and the same impact limiter design has
been retained. The weight increase for the package (cask and contents)
has been kept to a minimum consistent with the requirements. When
configured for the transport of HTGR fuel elements, the package weight
is 47,587 pounds and the maximum weight of the package when configured
for the transport of LWR fuel assemblies is 51,783 pounds. This is an
increase in weight of 4196 pounds or 8.8%.

2.2 Thermal Evaluation

A thermal analysis was performed for a FSV-1 cask as modified for shipping LWR spent fuel. The objective of the analysis was to determine the allowable fuel decay heat for a given cask surface temperature and the corresponding seal and canister temperatures. The following cask configurations regarding neutron shielding were considered: a cask with no additional shielding and a cask with 3 inches of external Boro-Silicone neutron shielding.

The curves (Figures 2.2-1 and 2.2-2) showing the allowable decay heat vs. the cask exterior temperature were generated assuming one-dimensional, steady-state heat flow and balancing the heat fluxes at the exterior surface of the cask.

Figure 2.2-1 is the plot of the exterior surface temperature of the cask, without additional shielding vs. the fuel heat load. The ambient temperature is $130^\circ F$, the insolation is 96 BTU/HR FT^2 (3314 watts solar heat load) and the surface emissivity is 0.5.

Figure 2.2-2 is the plot of the exterior surface temperature with 3 inches of Boro-Silicone neutron shielding added vs. the fuel heat load. The Boro-Silicone is covered with a stainless steel protective skin which has a surface emissivity of 0.5, the ambient temperature is $130^\circ F$ and the insolation is 96 BTU/HR FT^2 (4273 watts solar heat load).

Table 2.2-1 shows maximum canister temperature, the inner and outer seal temperatures and the fuel heat load for the different cask shielding concepts. The results are for the $130^\circ F$ day, an insolation of 96 BTU/HR FT^2 and a cask surface temperature of $180^\circ F$. The reported temperatures are for nominal internal gaps. The seal temperatures could vary approximately $\pm 5^\circ F$ depending on gap selection, and the canister temperature could vary approximately $\pm 40^\circ F$.

The external surface temperature of $180^\circ F$ (Ref. 49CFR173.393) was selected to establish the maximum fuel heat load that could be transported without the use of a personnel barrier to prevent access to the surface of the cask. An insolation of 96 BTU/HR FT^2 (Ref. GADR-55, Addendum 1) and an ambient temperature of $130^\circ F$ (Ref. USNRC Reg. Guide 7.8 were applied.

Fuel Heat Load vs Exterior Surface
Temperature with External Neutron Shielding

Solar Heat Load = 4273 Watts

Ambient temp. - 130 deg. F

Surface Emissivity - 0.5
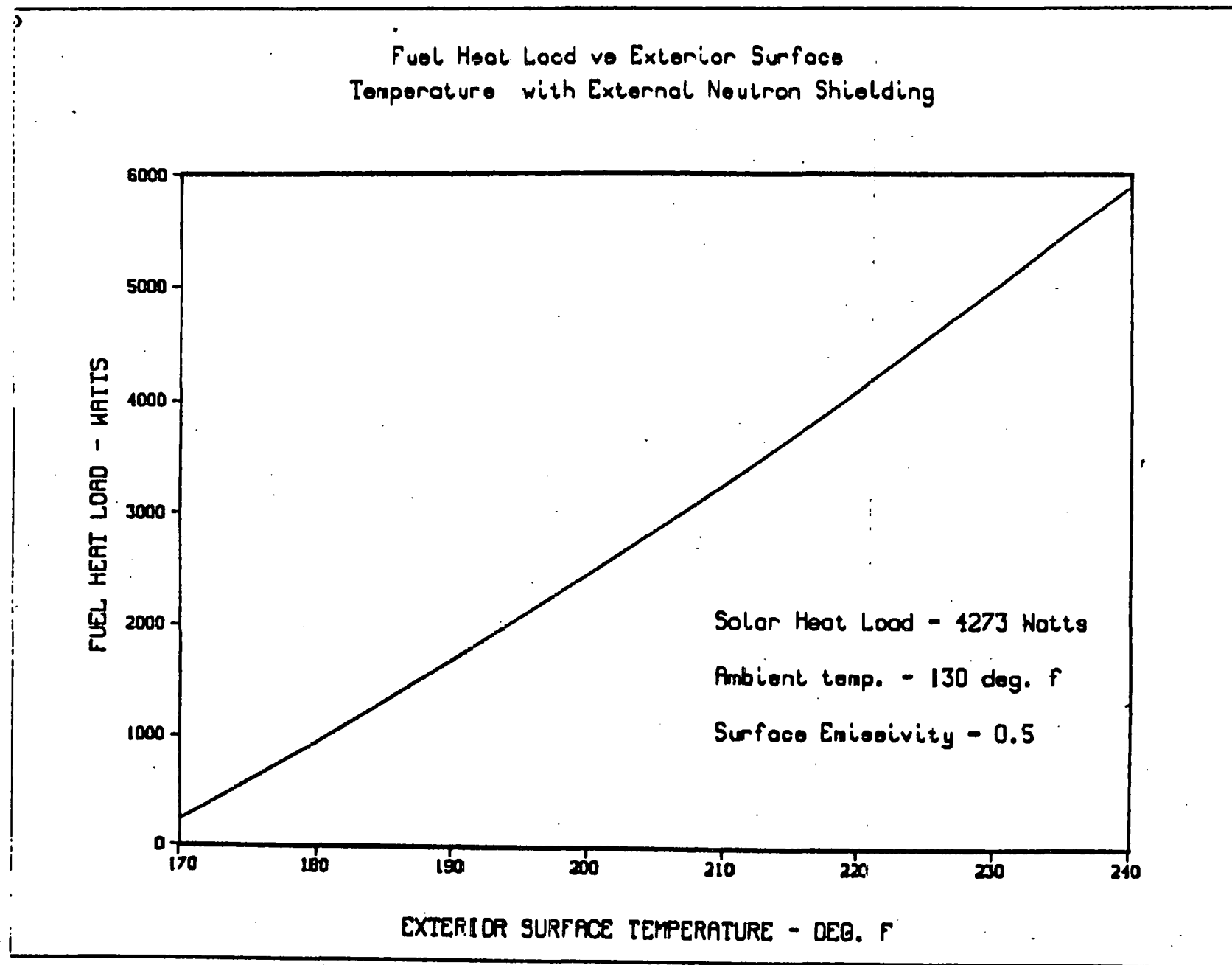
Figure 2.2-1

Fuel Heat Load vs Exterior Surface Temperature with No External Neutron Shielding

FUEL HEAT LOAD - WATTS

EXTERIOR SURFACE TEMPERATURE - DEG. F

Solar Heat Load = 3314 Watts

Ambient temp. = 130 deg. f

Surface Emissivity = 0.5
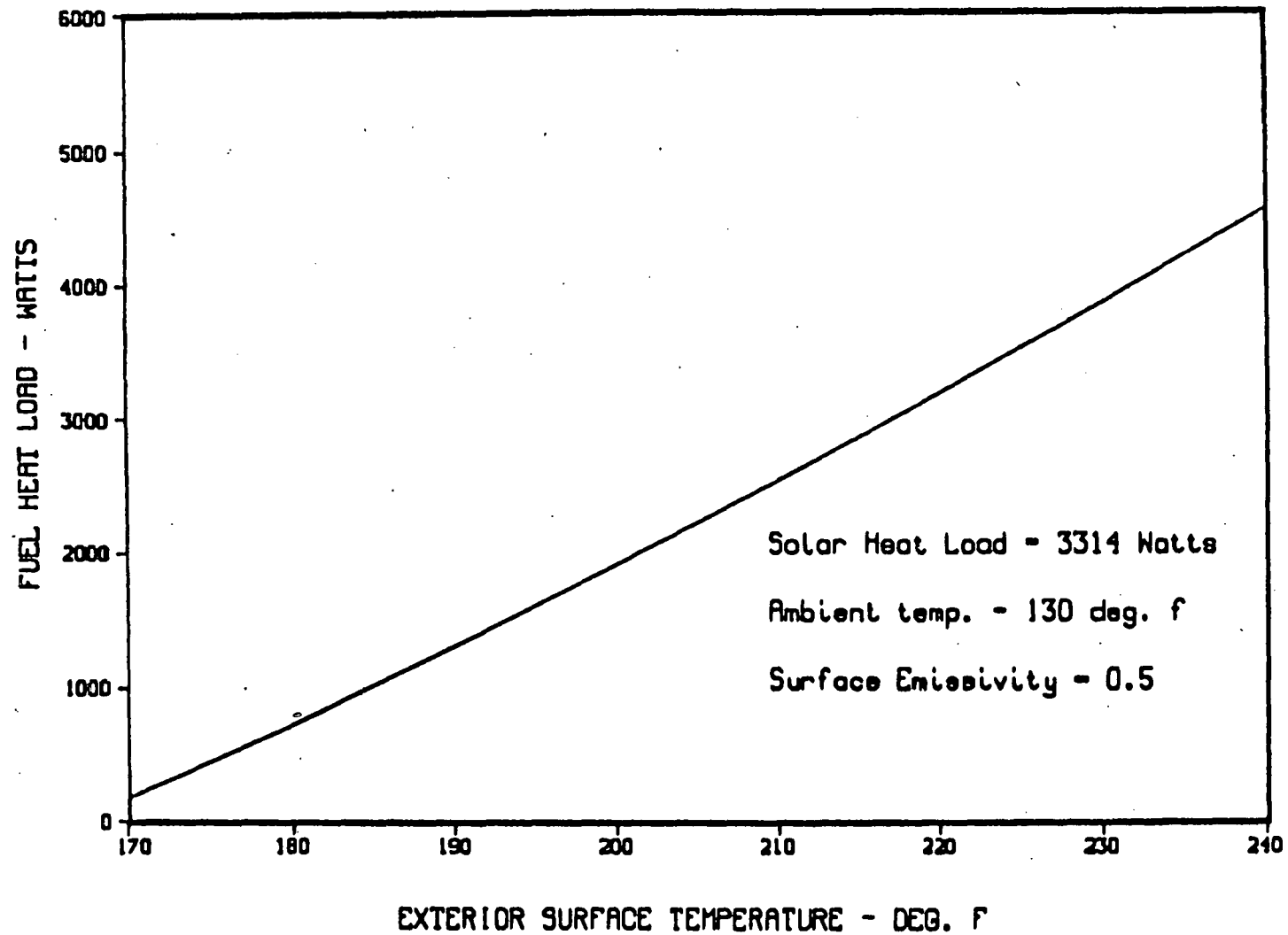
Figure 2.2-2

TABLE 2.2-1

CASK SEAL AND CANISTER TEMPERATURES

$(130^{\circ}F$ Ambient, 96 Btu/hr ft^2 Insolation
and a Cask Exterior Temperature of
$180^{\circ}F)$

	Canister ($^{\circ}$F)	First Seal ($^{\circ}$F)	Second Seal ($^{\circ}$F)	Fuel Heat Load (watts)
Unshielded Cask	264	184	184	830
Boro-Silicone external shielding	281	200	200	895

Note: The USNRC Certificate of Compliance (6346 Rev 3) for the FSV-1 cask allows for contents with a maximum fuel heat load of 4.1 kilowatts. By allowing the exterior temperature to increase above $180^{\circ}F$ the fuel heat load can be increased to the 4100 watt limit without exceeding the allowable seal temperature of $400^{\circ}F$.

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Using this criteria the allowable fuel heat load is approximately 895 watts for the cask with neutron shielding and 830 watts for the cask without neutron shielding. These limits would not be very restrictive for PWR fuel assemblies after 5 years decay but would be restrictive for remaining spectrum of LWR fuel assemblies. Based on this analysis a personnel barrier will be included in the modifications to the FSV-1 cask, however, the surface temperature of the cask will vary nearly directly as the ambient air temperature. Thus when the ambient temperature is expected to be 100°F or lower, all but a few LWR fuel assemblies with the highest heat loads can be transported without the use of the personnel barrier.

2.3 Shielding Evaluation

The model FSV-1 cask was evaluated for suitability for the shipment of Light Water Reactor (LWR) fuel; either one Pressurized Water Reactor (PWR) or three Boiling Water Reactor (BWR) fuel assemblies. Radiation dose rates were calculated based on the existing cask shielding and for several cases of additional gamma and neutron shielding. Both PWR and BWR fuels with a range of burnup and decay times were considered as the radiation source. Although several options of supplemental gamma and neutron shielding were analyzed, the addition of 3 inches of Boro-Silicone neutron shielding to the exterior of the FSV-1 cask has been selected as the optimum for added shielding.

The Department of Transportation (DOT) external radiation dose rate (per 49CFR173.393) limit of 10 mrem/hr at 6 feet from the edge of the transport vehicle was used to determine which fuel assemblies could be transported. Based on this determination the maximum burnup in Megawatt Days/Metric Ton Heavy Metal (MWD/MTHM) and the associated minimum decay time for each fuel type are:

Transportable with No Supplemental Shielding

Type	Quantity	Burnup (MD/MTHM)	Decay (years)
PWR	1	15,000	1
PWR	1	23,000	2
PWR	1	30,000	5
PWR	1	32,000	10
BWR	3	–	1
BWR	3	12,500	2
BWR	3	18,000	5
BWR	3	25,000	10

Transportable with 3 Inches of Boro-Silicone Neutron Shielding

Type	Quantity	Burnup (MD/MTHM)	Decay (years)
PWR	1	26,000	1
PWR	1	37,000	2
PWR	1	45,000	5
PWR	1	>50,000	10
BWR	3	–	1
BWR	3	42,000	2
BWR	3	43,000	5
BWR	3	>50,000	10

Note: The values in this table have been revised to reduce the SANDIA-ORIGEN (α,n) conversion by a factor of 2 and zero out the 2.75 MeV group.

The trade-off between burnup and decay for no additional shielding and for 3 inches of Boro-Silicone neutron shielding is graphically displayed in Figure 2.3-1.

The information presented in this summary of the shielding evaluation has been revised to include a correction for the gamma decay and α,η conversion in the SANDIA-ORIGEN code. The data reported in the balance of the shielding evaluation is uncorrected. Due to the nature of the error in the reference code the uncorrected results are conservative, i.e.: LWR fuel assemblies with higher burnup and shorter decay time can be transported without exceeding the 10 mrem/hr at 6 feet criteria.

2.4 Containment Evaluation

During the transport of HTGR fuel elements in the FSV-1 cask, the primary containment boundary is provided by the inner container (reference 7.2). This inner container closure has a seal assembly with two concentric, silicone rubber seals. This arrangement provides for a seal interspace which is evacuated and the pressure rise monitored to verify an acceptable leakage rate prior to each use. The outer closure which seals to the cask body has a similar seal arrangement, which is tested for an acceptable · leakage rate as part of the annual testing and maintenance program for the FSV-I cask.

For this evaluation of the suitability of the cask for the transport of LWR irradiated fuel assemblies, the inner container is removed from the FSV-1 cask to provide a larger cavity, and thus accommodate larger canisters. Without the inner container, the regulatory requirements for primary containment must be satisfied by other components of the shipping package, i.e. cask and/or contents.

There are four possibilities for providing the required containment during both the normal and accident conditions of transport:

Case 1 - Design the modified cask closure seals to satisfy the regulatory requirements concerning leakage rates for both normal and accident conditions of transport.

Case 2 - Design the LWR fuel canister to satisfy the regulatory requirements for "special form" material.

Case 3 - Design the canister as a "containment vessel" and design the cask body to protect this "containment vessel".

Case 4 - Design the canister as a "containment vessel" for normal conditions of transport and design the cask body closure to provide containment during accident conditions of transport.

For Case 1, the USNRC Regulatory Guide 7.4 "Leakage Tests on Packages for Shipment of Radioactive Materials", provides guidance to satisfy the requirements of 10CFR71 by establishing leakage rates and leakage tests in accordance with ANSI N14.5, "Draft American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials."

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The mechanical integrity of the cladding on LWR fuel elements should be adequate to provide containment of the pellets and the gases during the normal conditions of transport and could possibly provide containment during the accident conditions. The design and use of impact limiters probably controls whether the LWR fuel element cladding can survive the impact portion of the accident conditions of transport. Since the FSV-1 cask does not use an impact limiter on the bottom of the cask, the survivability of the fuel element cladding is doubtful, and the closure seals will need to satisfy the leakage rates allowable for accident conditions of transport.

For Case 2 the regulatory requirements are imposed by 10CFR71, subpart A, section 71.4 (0) (2) as follows:

"(0) "Special form" means any of the following physical forms of licensed material of any transport group:

(2) The material is securely contained in a capsule having no dimension less than 0.5 millimeter or at least one dimension greater than five millimeters, which will retain its contents if subjected to the tests prescribed in Appendix D of this part; and which is constructed of materials which do not melt, sublime, or ignite in air at 1,475°F., and do not dissolve or convert into dispersible form to the extent of more than 0.005 percent by weight by immersion for 1 week in water at 68°F. or in air at 86°F."

10CFR, Part 71, APPENDIX D - TESTS FOR SPECIAL FORM LICENSED MATERIAL

"1. Free Drop - A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in such a position as to suffer maximum damage.

2. Percussion - Impact of the flat circular end of a 1 inch diameter steel rod weighing 3 pounds, dropped through a distance of 40 inches. The capsule or material shall be placed on a sheet of lead, of hardness number 3.5 to 4.5 on the Vickers scale, and not more than 1 inch thick, supported by a smooth essentially unyielding surface.

3. Heating - Heating in air to a temperature of 1,475°F. and remaining at that temperature for a period of 10 minutes.

4. Immersion - Immersion for 24 hours in water at room temperature. The water shall be at pH 6-pH 8, with a maximum conductivity of 10 micromhos per centimeter."

If the canister is qualified as "special form", the only containment concern will be the air or gas that occupies the space inside the cask and outside of the canister. This air or gas will not be "primary coolant" as defined in 10 CFR 71 but rather part of the authorized contents of the cask. If this air or gas contains no radioactivity, the closure seal can be considered a secondary seal. If the air or gas contains radioactivity, the leakage rate and leakage test determined per ANSI N14.5 will be applied to the inner closure seals.

For Case 3, the canister provides the containment as in case 2, however, does not qualify as "special form", but is protected by the cask body during the normal and accident conditions of transport. This "containment vessel" must be leakage tested per USNRC Regulatory Guide 7.4, prior to each shipment.

For Case 4 the canister functions as a "containment vessel" during the normal conditions of transport but is not designed nor protected by the cask body such that it survives the accident conditions of transport. The cask closure seals must, for this case, be leakage tested per USNRC Regulatory Guide 7.4, using ANSI N14.5 to establish allowable leakage rates for the accident conditions of transport, prior to each shipment. This case is very similar to the LWR irradiated fuel shipping casks that can depend on the fuel rod cladding for containment during the normal conditions of transport.

The present design concept for the FSV-1 cask, modified for the transport of LWR irradiated fuel assemblies does provide for a seal assembly as part of the inner closure. This seal assembly can be leakage tested and thus satisfy the requirements for a containment boundary. When the canister design is finalized, and its contribution toward satisfying primary containment is evaluated, the cask closure design and the cask handling (leakage testing) procedures can then be optimized.

2.5 Discussion of License Evaluation

The final test of the suitability of the FSV-1 cask, appropriately modified, for the transport of LWR irradiated fuel assemblies will be the issuance of a revision to the Certificate of Compliance, which adds the LWR fuel the the authorized contents.

During this evaluation, considerable care has been used in the selection of the neutron shielding material, in the selection of the candidate LWR irradiated fuel assemblies and the design modifications, to limit the amount of analysis that will be required to support the request for a revision to the Certificate of Compliance.

Structural load paths for the modified design are the same or equivalent to the existing design. The weight increase in the package has been limited to slightly less than 9%. Further structural analysis will be required to determine if the existing cask structure can accept this increase in weight. For most load conditions the applied stress levels are sufficiently below the allowable stress levels to accommodate this increase in the loads.

The current Certificate of Compliance authorizes contents with a total heat load of 4.1 kilowatts. Some of the LWR irradiated fuel assemblies selected for transport in the modified FSV-1 cask could have a heat load slightly greater than 4.1 kw. The existing fuel heat load limit is based on the expected decay heat from 6 irradiated HTGR fuel elements after 100 days decay rather than any finite capacity of the FSV-1 cask. If it is desirable to ship the few LWR fuel assemblies with decay heat in excess of 4.1 kw, the appropriate revision to the Certificate of Compliance appears feasible.

The spectrum of LWR irradiated fuel assemblies that can be transported in the modified FSV-1 cask will be limited by the neutron and gamma shielding. Specific radiation levels, external to the cask, are not a part of the Certificate of Compliance, however, the regulatory requirement of 49CFR173.393 must be satisfied prior to each shipment. Therefore, that spectrum of LWR fuel assemblies that will not exceed the allowable radiation levels, should be identified in the authorized contents.

Since the containment boudary for the FSV-1 cask, when configured for the transport of HTGR fuel, is provided by the inner container, removing this container during the transport of LWR fuel assemblies will require that another containment boundary be identified in the revised Certificate of Compliance. Containment of radioactive material within a shipping package (cask and contents) may be provided by the cask or the contents or some combination of both, i.e. the contents may provide containment for normal conditions of transport and the cask may provide containment for accident conditions of transport.

The design concept for the modified inner closure for the FSV-1 has provisions for a seal that can be leakage tested and thus satisfy the requirements for a containment boundary. There are two existing penetrations in the base of the cask body that will become part of the containment boundary when the inner container is removed. Although the seals associated with these penetrations are not easily leakage tested, these penetrations are not opened and closed when the cask is loaded or unloaded and thus the seals are not disturbed and a periodic leakage test may be justifiable.

3.0 SHIELDING STUDY

3.1 INTRODUCTION

3.1.1 Design Shielding Analysis for HTGR Fuel

The Final Design Report for the FSV-1 fuel shipping cask (Ref. 3-1) describes the engineering analyses which led to the final design shown in Reference 6.2. It covers the physical dimensions and materials of construction, weight calculations, stress analysis, shielding analysis, heat transfer analysis, criticality analysis, and several other facets of the overall design.

The original shielding analysis for the FSV-1 cask was performed in 1967 utilizing the PATH gamma shielding code (Ref. 3-2). At that time, little was known about the possible emission of neutrons from spent fuel, but fortunately later studies (Ref. 3-3) showed that the neutron emission from HTGR spent fuel was minimal because of the numerous capture events needed to transmute TH-232 or U-233 to Cm-242 or Cm-244. Hence, the design of the FSV-1 cask with only stainless steel and depleted uranium gamma shielding proved to be the correct approach for the transport of HTGR spent fuel.

In 1977 and 1978 the closure system for the FSV cask was reanalyzed and modified in order to reduce the external radiation levels during the cask handling procedures. Details of this modification are reported in an addendum to the original design report. (Ref. 3-4).

Finally, in 1979 some actual dose-rate measurements became available when the cask was used to transport a surveillance spent fuel element from Fort St. Vrain to GA. The calculated gamma dose rates on the surface of the cask were within a factor of two of the measured dose rates (Ref. 3-5), and hence, the shielding design of the cask for the transport of HTGR fuel elements is essentially as claimed in the design report.

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3.1.2 Earlier LWR Application Studies

Possible use of the FSV-1 cask for the shipment of LWR spent fuel was first considered in 1971 (Ref. 3-6). It was recognized at this time that some form of neutron shielding might have to be added in order to accommodate LWR fuel with appreciable burnup. A series of independent studies on LWR shipping cask concepts (Ref. 3-7) confirmed this conclusion.

Further studies were performed on shipping cask applications in 1978 (Ref. 3-8). It was assumed that the FSV-1 cask could accommodate one PWR assembly or four BWR assemblies, and that only neutron shielding could be added to the cask. The preliminary results from this 1978 study are shown in Table 3-1.

As will be shown, these earlier results, especially for the PWR, did correctly predict the trend of neutron shield requirements for the FSV-1 cask. However, a much more accurate analysis was needed, utilizing up-dated source terms and adequate computer calculations.

3.1.3 Purpose of Present Shielding Study

The purpose of the present shielding study on the FSV-1 cask is to determine with a good degree of certainty its adequacy to transport LWR spent fuel elements with or without additional shielding. It differs from previous studies in making more extensive use of discrete ordinates transport calculations and utilizing more timely and complete input information on source terms and fuel element configuration. The planned steps in this study included:

1. Gather input information (burnup, specific power, power distribution, decay time, source terms, geometry, composition, container details, changes in dose rate criteria if any, etc.)

2. Calculate side neutron dose rates as function of neutron shielding thickness and location using the 1-D DTFX transport code. Also calculate secondary fissions and capture gamma production.

TABLE 3-1

SUMMARY OF 1978 RESULTS - LWR FUEL WHICH COULD BE SHIPPED IN FSV-1 CASK

	Burnup (MWD/MTHM)	
	PWR	BWR
No Additional Shielding		
Decay Time (yr)		
1	Unfeasible	Unfeasible
2	20,000	18,000
3	23,500	22,000
10	26,500	24,500
4-in. Neutron Shielding Added		
Decay Time (yr)		
1	Unfeasible	Unfeasible
2	37,500	37,000
3	45,000	41,500
10	>50,000	50,000

3. Calculate side gamma dose rates as function of neutron shielding thickness and location , including primary and secondary sources, using the PATH gamma-shielding code or DTFX transport code.

4. Create parametric curves from 2 and 3 to permit a selection of neutron/gamma split and appropriate neutron shield thickness. Include approximate corrections for ground scattering.

5. Perform dose rate calculations at ends of cask.

This shielding writeup will cover the methodology and calculational approach, source terms, geometry and compositions, and results. Presentation of the results has been carefully tailored to provide the quickest answer to the question "How can the FSV-1 best carry LWR fuel of X burnup and Y decay time?"

3.2 METHODOLOGY AND CALCULATIONAL APPROACH

3.2.1 Design Criteria

The maximum permissible radiation dose rates during transport are established by Title 49, Code of Federal Regulations, Part 173,393. The specific limit used for this study is: 10 mrem/hr at a distance of 6 feet from the edge of the transport vehicle. Design guidance from the NWTS Test Program further limits the radiation dose rate for contact operations to a maximum of 50 mrem/hr. This limit of 50 mrem/hr is applicable to the outer surface of the inner closure prior to the installation of the outer closure.

3.2.2 Direct Neutron Calculations

Direct neutron calculations were performed using the one-dimensional transport code DTFX (Ref. 3-20). This code numerically solves (by the method of discrete ordinates) the multigroup form of the neutron transport equation in one-dimensional plane, cylindrical, and spherical geometries. Due to the long cylindrical geometries involved, one-dimensional analysis is more than adequate. The DTFX code just described was chosen because of its calculational method and simplicity of use.

The neutron flux-to-dose-rate conversion factors used in the DTFX calculations were obtained from Ref. 3-9 and are shown in Figs. 3-1 and 3-2 for neutron energies from 10^{-8} to 10^2 MeV. Table 3-2 gives the conversion factors for the energy ranges used in DTFX.

Neutron calculations were performed using a neutron spectrum normalized to 1 n/cm^3-sec. The results of the DTFX computer calculation were then multiplied by the appropriate neutron source strength for a fuel element with a specific burnup and decay time. The cylindrical calculations were run in the P_3, S_8 application. Slab geometry was utilized with S_{16} quadrature for the top and bottom dose rate calculation with correction factors applied to the results to correct for the actual finite slab geometry.

The DTFX neutron calculations were run for two cases: (1) the FSV-1 cask without additional shielding and (2) the FSV-1 cask with 3 in. of nuetron shield added to the outside. Both of these cases considered no additional gamma shield. In order for parametric studies to be analyzed, Figs, 3-3 and 3-4 were utilized for neutron dose rate generation. For example, the neutron data for no neutron shield and 1 in. gamma shield is generated by multiplying the data for no neutron and no gamma shield by the 1 in. factor in Fig. 3-3. The same method is applied to all other studies.

3.2.3 Direct Gamma Calculations

Direct gamma calculations were accomplished with the help of the point-kernel computer code PATH (Ref. 3-2). This code is based upon exponential attenuation from source point to dose point, adjusted by appropriate buildup factors. The flux-to-dose-rate conversion factors were adjusted to comply with the ANS Standard 6.1.1 (Ref. 3-9). This data standard is shown in Fig. 3-5 for gammas with energies between 10^{-2} and 10^2 MeV. Table 3-3 tabulates these factors for 38 discrete photon energies. Corrections for multilayer buildup were also applied, since the calculations performed by most point-kernel codes take into account only one-layer buildup factors. Corrections for two-layer configurations of iron and uranium (Ref. 3-10) were applied when deemed appropriate.
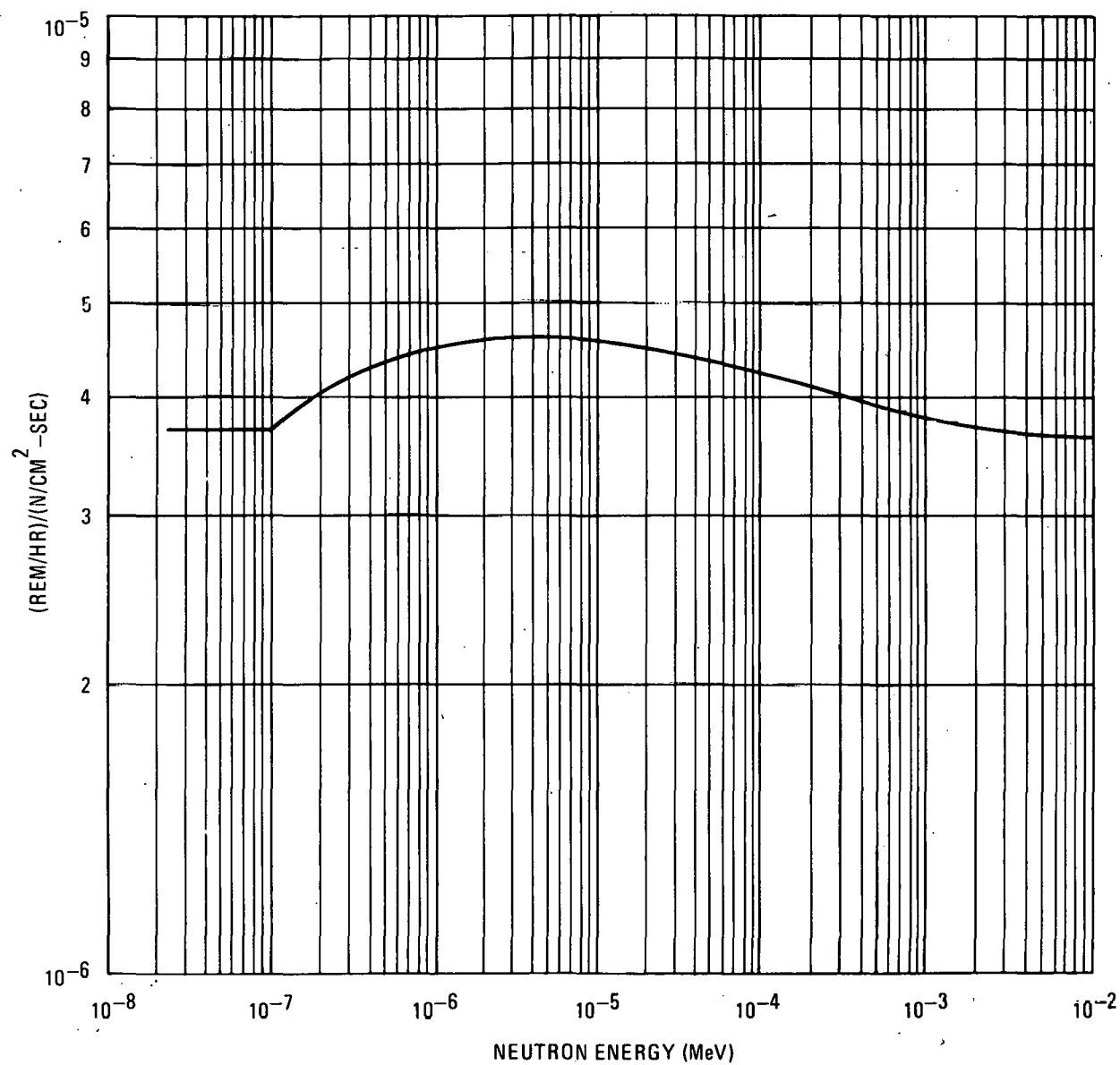
Fig. 3.1.   Neutron dose-to-flux conversion factors
vs. neutron energy

26
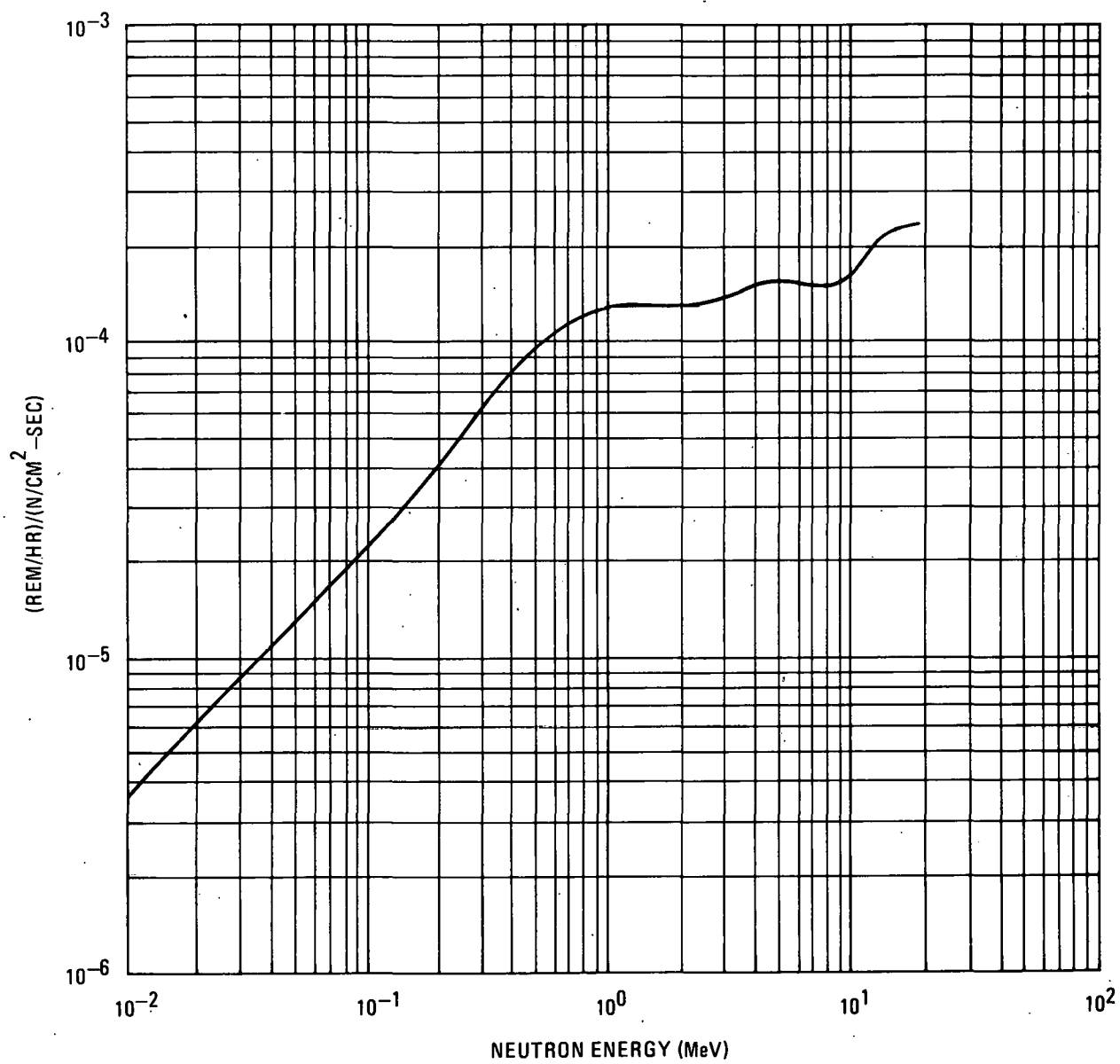
Fig. 3-2. Neutron dose-to-flux conversion factors
vs. neutron energy

TABLE 3-2
SOURCE INPUT AND DOSE CONVERSION FACTORS FOR ORNL 36 GROUPS

Group[a]	Energy Range	Midpoint Energy	Conversion Factor $\left(\dfrac{\text{rem/hr}}{\text{Particles/cm}^2\text{-sec}}\right)$
1	14.9–6.70 MeV	10.8 MeV	1.6-4
2	6.70–3.68	5.19	1.5-4
3	3.68–2.02	2.85	1.3-4
4	2.02–1.35	1.68	1.3-4
5	1.35–0.907	1.13	1.3-4
6	0.907–0.498	0.7	1.1-4
7	0.498–0.334	0.42	8.1-5
8	0.334–0.183	0.26	5.2-5
9	0.183–0.123	0.15	3.0-5
10	123–52.5 KeV	87.8 KeV	2.0-5
11	52.5–40.9	46.9	1.2-5
12	40.9–24.8	32.9	9.1-6
13	24.8–15.0	19.9	6.0-6
14	15.0–7.10	11.0	3.8-6
15	7.10–3.35	5.2	3.6-6
16	3.35–1.23	2.3	3.7-6
17	1.23–0.454	0.84	3.8-6
18	0.454–0.101	0.28	4.0-6
19	101–47.9 eV	74.5 eV	4.2-6
20	47.9–22.6	35.3	4.3-6
21	22.6–10.7	16.7	4.4-6
22	10.7–2.38	6.5	4.5-6
23	2.38–1.13	1.8	4.4-6
24	1.13–0.414	0.77	4.3-6
25	0.414–0.1	0.26	4.1-6
26	0.1–0.0	0.05	3.7-6
27	13–8 MeV	10.5 MeV	9.5-6
28	8–4.92	6.5	6.9-6
29	4.92–3.03	4.0	5.0-6
30	3.03–1.86	2.5	3.6-6
31	1.86–1.15	1.5	2.7-6
32	1.15–0.705	0.93	1.9-6
33	0.705–0.34	0.53	1.2-6
34	0.34–0.129	0.24	5.9-7
35	0.129–0.049	0.09	2.8-7
36	0.049–0.03	0.04	4.4-7

[a] Groups 1-26 for neutrons and groups 27-36 for gammas.
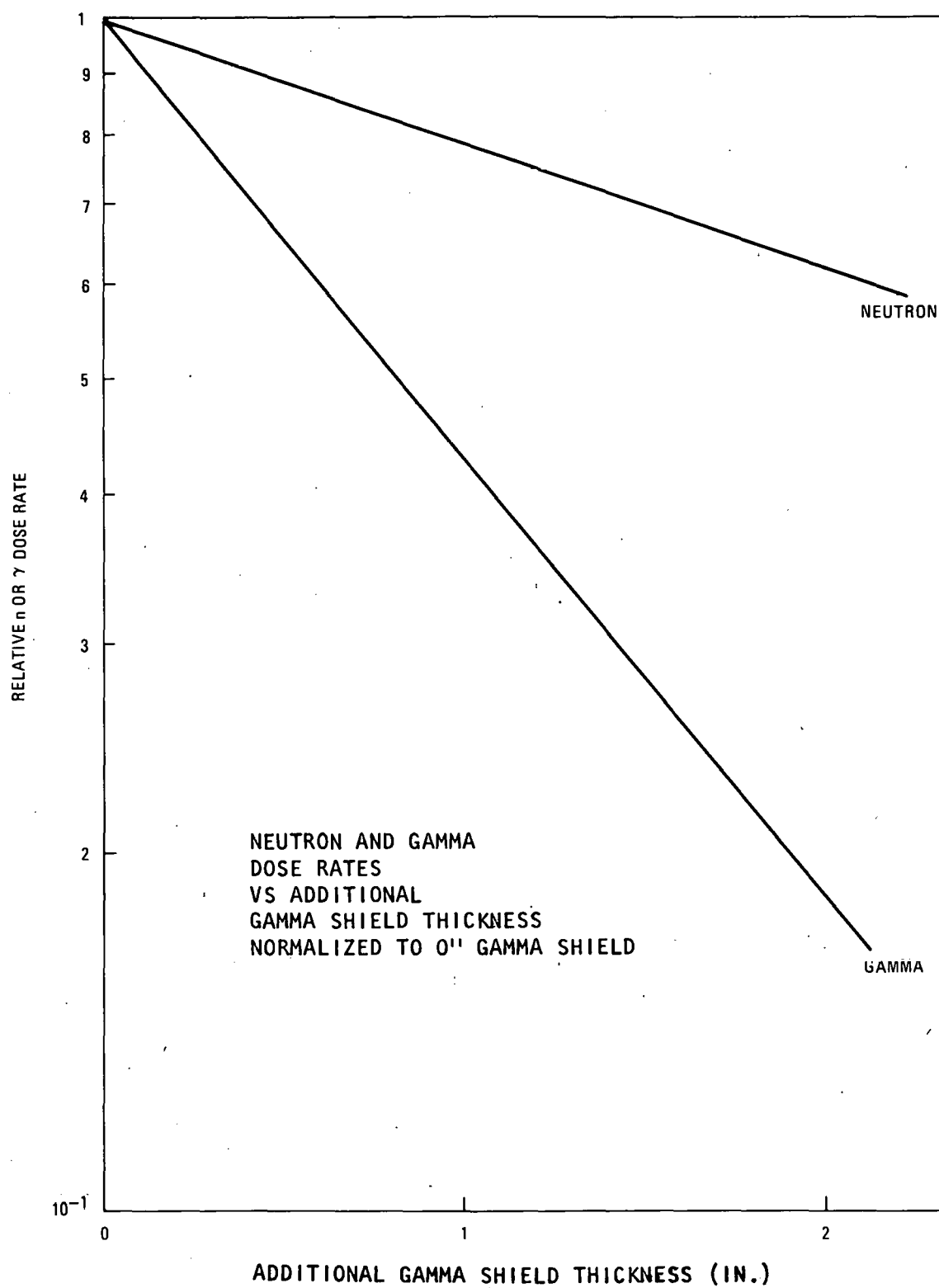
Fig. 3-3. Neutron and gamma dose rates vs additional gamma shield
           thickness normalized to 0" additional gamma shielding
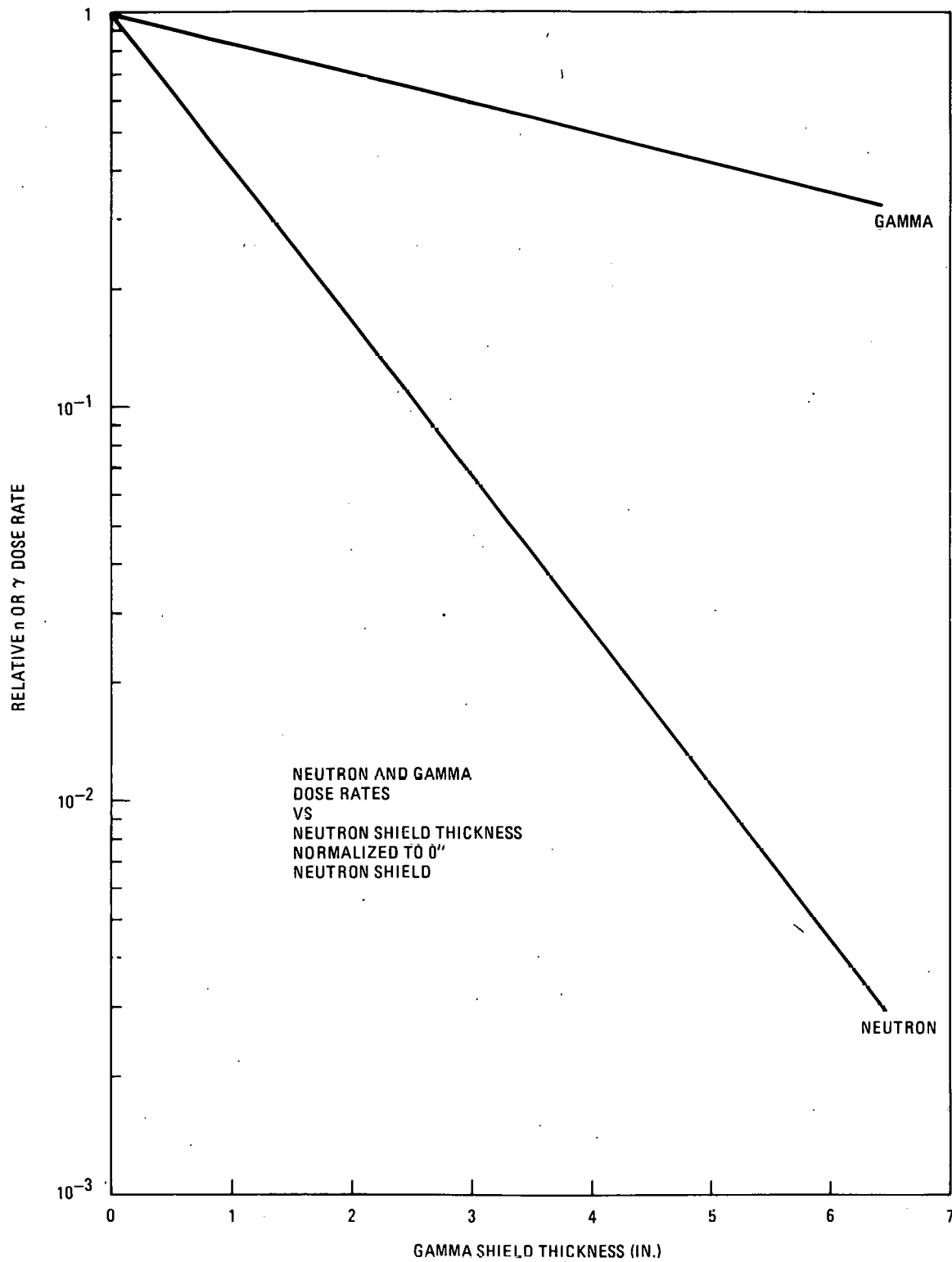
Fig. 3-4. Neutron and gamma dose rates vs neutron
shield thickness normalized to 0"
neutron shield

30
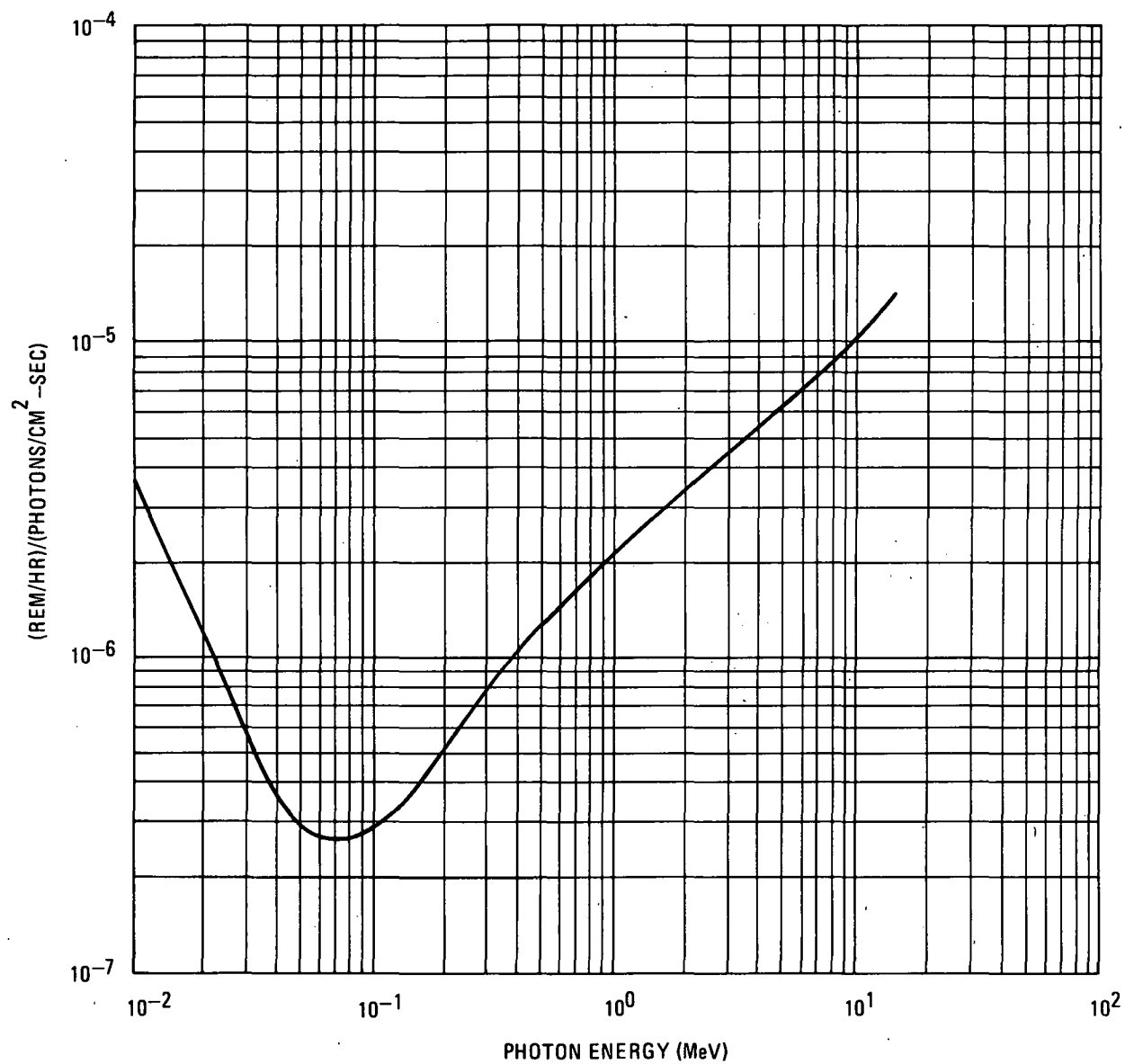
Fig. 3-5. Photon dose-to-flux conversion factors vs. photon energy

TABLE 3-3
GAMMA-RAY FLUX-TO-DOSE-RATE CONVERSION FACTORS

Photon Energy (MeV)	(rem/hr)/ (photon/cm^2-sec)
0.01	3.96-06
0.03	5.82-07
0.05	2.90-07
0.07	2.58-07
0.1	2.83-07
0.15	3.79-07
0.2	5.01-07
0.25	6.31-07
0.3	7.59-07
0.35	8.78=07
0.4	9.85-07
0.45	1.08-06
0.5	1.17-06
0.55	1.27-06
0.6	1.36-06
0.65	1.44-06
0.7	1.52-06
0.8	1.68-06
1.0	1.98-06
1.4	2.51-06
1.8	2.99-06
2.2	3.42-06
2.6	3.82-06
2.8	4.01-06
3.25	4.41-06
3.75	4.83-06
4.25	5.23-06
4.75	5.60-06
5.0	5.80-06
5.25	6.01-06
5.75	6.38-06
6.25	6.74=06
6.75	7.11-06
7.5	7.66=06
9.0	8.77-06
11.0	1.03-05
13.0	1.18-05
15.0	1.33-05

All PATH gamma-ray calculations for PWR fuel were run assuming that the fuel element burnup was 20,000 MWD/MTHM with a decay time of 1 yr. Calculational results for the remaining cases of burnup and decay time were obtained using the following equation:

$$DR(B,t) = \sum_{i=1}^{6} DR_i(20000,1) \times \frac{S_i(B,t)}{S_i(20000,1)}$$

where $DR(B,t)$ = total dose rate due to a fuel element with
                burnup B and decay time t, R/hr,

      i = gamma energy group index,

   $DR_i(20000,1)$ = dose rate due to energy group i with burnup
                equal to 20,000 MWD/MTHM and decay time equal
                to 1 yr, R/hr,

      $S_i(B,t)$= source strength of energy group i for a specific
                burnup B and decay time t, photons/cm^3-sec.,

   $S_i(20000,1)$ = source strength of energy group i for a burnup
                of 20,000 MWD/MTHM and decay time 1 yr, photons/cm^3-sec.

PATH gamma calculations off the side of the cask considered 0, 3, and 6-in. of neutron shield added to the outside. Parametric studies involving extra added gamma shield were calculated the same way as the neutron studies explained earlier. Dose point locations were taken 6 ft. from the edge of the transport vehicle (119 in. from center of cask). Gamma calculations for the rear of the cask considered dose point locations on the bottom surface of the cask and 6 ft. from the rear of the transport vehicle (154 in. from bottom of cask).

3.2.4 Ground Scattering

The effect of ground scattering of radiation has been widely calculated and reported in the literature (for instance, see Ref. 3-11). Much of this work was done in connection with the Aircraft Nuclear Propulsion program, the space program, and the civil defense program.

The ground-scattered dose rate from the NL 10/24 cask was calculated in some detail by NL Industries (Ref. 3-12). For the normal case of 6 ft. from the edge of the transport vehicle, the ground-scattered neutron dose rate was 40% of the direct neutron dose rate, while the ground-scattered gamma dose rate was 2.3% of the direct gamma dose rate. Other investigators have found somewhat higher gamma scattering percentages (Ref. 3-11). For this shielding study, the correction factors for ground scatter were taken as 1.4 for neutrons and 1.1 for gammas.

3.3 SOURCE TERMS

Neutron and gamma source terms were supplied by Sandia Laboratories, utilizing the SANDIA-ORIGEN computer code (Ref. 3-13). ORIGEN runs were made for the following four cases:

Case 1: PWR fuel assembly, 20,000 MWD/MTHM burnup, 37.3 MW/MTHM specific power, 3.3% enrichment, 80% load factor, fuel removed from core just before end of second year in reactor.

Case 2: PWR fuel assembly, 33000 MWD/MTHM burnup, other quantities same as Case 1, fuel removed from core after three years.

Case 3: PWR fuel assembly, 40,000 MWD/MTHM burnup, 45.2 MW/MTHM specific power, other quantities same as Case 2.

Case 4: BWR fuel assembly, 34,000 MWD/MTHM burnup, 2.66% enrichment, 80% load factor, fuel removed after four years, variable specific power as follows:

33.3 MW/MTHM	1st burnup phase
32.0 MW/MTHM	2nd burnup phase
29.4 MW/MTHM	3rd burnup phase
28.8 MW/MTHM	4th burnup phase

ORIGEN, with its extensive and impressive printout, is only as good as its stored nuclear data. An effort will be made in the near future to identify the differences between ORIGEN-1, SANDIA-ORIGEN, and ORIGEN-2. In the meantime, a general discussion of the source terms and their relative importance will have to suffice.

3.3.1 Fission Products

SANDIA-ORIGEN calculates the inventories of 708 fission products. The flux and burnup calculations utilize only three neutron-energy groups (fast, resonance, thermal). These three groups are collapsed into a single group for the actual burnup calculations. Presumably, the cross section averaging is done for an average mix of fuel, moderator, fission products, actinides, poison, and structure at an average temperature. As certain high-cross-section products build up, it is obvious that the single-group cross sections will become more and more approximate (as self-shielding and flux spectrum changes occur). Therefore, uncertainties in the fission-product inventories -- especially for high burnup -- are to be expected.

The ORIGEN code prints out a complete nuclide listing of fission products as a function of decay time, in units of gram atoms, grams, and curies. These quantities are then converted to watts of decay heat and MeV/sec of gamma emission (by photon energy group). In Table A-1 of the appendix, the gamma source strengths for the three PWR cases and the single BWR case are reproduced directly from the SANDIA-ORIGEN runs.

There is no direct way to clearly identify which nuclides are contributing the most to the shielded dose rate, since the nuclide emissions are summed together. However, as will be shown later, the 1.99- and 2.75-MeV photon groups are the most important, thus strongly suggesting that Pr-144 is a major contributor to the shielded gamma dose rate at one to three years' decay. The 2.75 MeV energy group contributor was found to be (erroneously) Ag-110m, a matter which will be discussed later. In any follow-on study, the fission yields, absorption cross sections, and emission spectra being used in ORIGEN for Ce/Pr-144 and Ru/Rh-106 should be carefully evaluated.

3.3.2 Actinides

The actinides of most importance to the FSV-1 cask shielding study are Pu-238, Cm-242, and Cm-244, all of which emit neutrons by spontaneous fission and/or alpha-n reactions. The ORIGEN code tabulates actinides by

gram-atoms, grams, and curies, along with conversions to watts and n/sec.
The actinide listing is a little more helpful than the fission product
listing in that neutron emission rates are given by nuclide.. For example,
the CM-244 emission rates are 2.38+3 (α,n) neutrons/Ci-sec and 1.41+5 spon-
taneous fission neutrons/Ci-sec. The (α,n) yield is approximately twice
that calculated for UO_2 (Ref. 3-14). In the case of Cm-242, the (α,n) yield
used in ORIGEN is 2910 n/Ci-sec, a factor of 2.5 higher. Therefore, the
neutron source terms (as listed in Table A2 of the appendix) are probably
somewhat conservative. The relative yields of Cm242 and Cm244 are shown
graphically in Figs. 3-6 to 3-9.

3.3.3 Activitation Products

The ORIGEN code also calculates activation products (i.e. structural
materials; activitated fission products such as Cs-134 are considered under
fission products). This information is not particularly relevant to this
study for three reasons:

1. Most of the fuel assembly structure is located at either end
    of the assembly, but in the SANDIA-ORIGEN runs it is homogenized
    with the fuel. Thus, the structural activitation levels are not
    necessarily correct.
2. Stainless steel was assumed to be completely free of cobalt.
3. In any event, gamma emission from the structural materials is
    negligible compared with that from the fission products (compare
    Table A-3 with Table A-1 in the appendix).

3.3.4 Input Data

The neutron source strength data used as input for DTFX is due to
the actinides present in the fuel assembly. Table 3-4 and Fig. 3-10 give
the neutron source strengths for different PWR and BWR burnups as a function
of decay time. The neutron source spectrum was also obtained from Ref. 3-12.
Table 3-5 shows the spectrum collapsed into the DTFX input energy groups
and normalized to 1 neutron/cm^3-sec. This source spectrum is also shown in
Fig. 3-11 and agrees quite well with previously published work (Ref.3-15).

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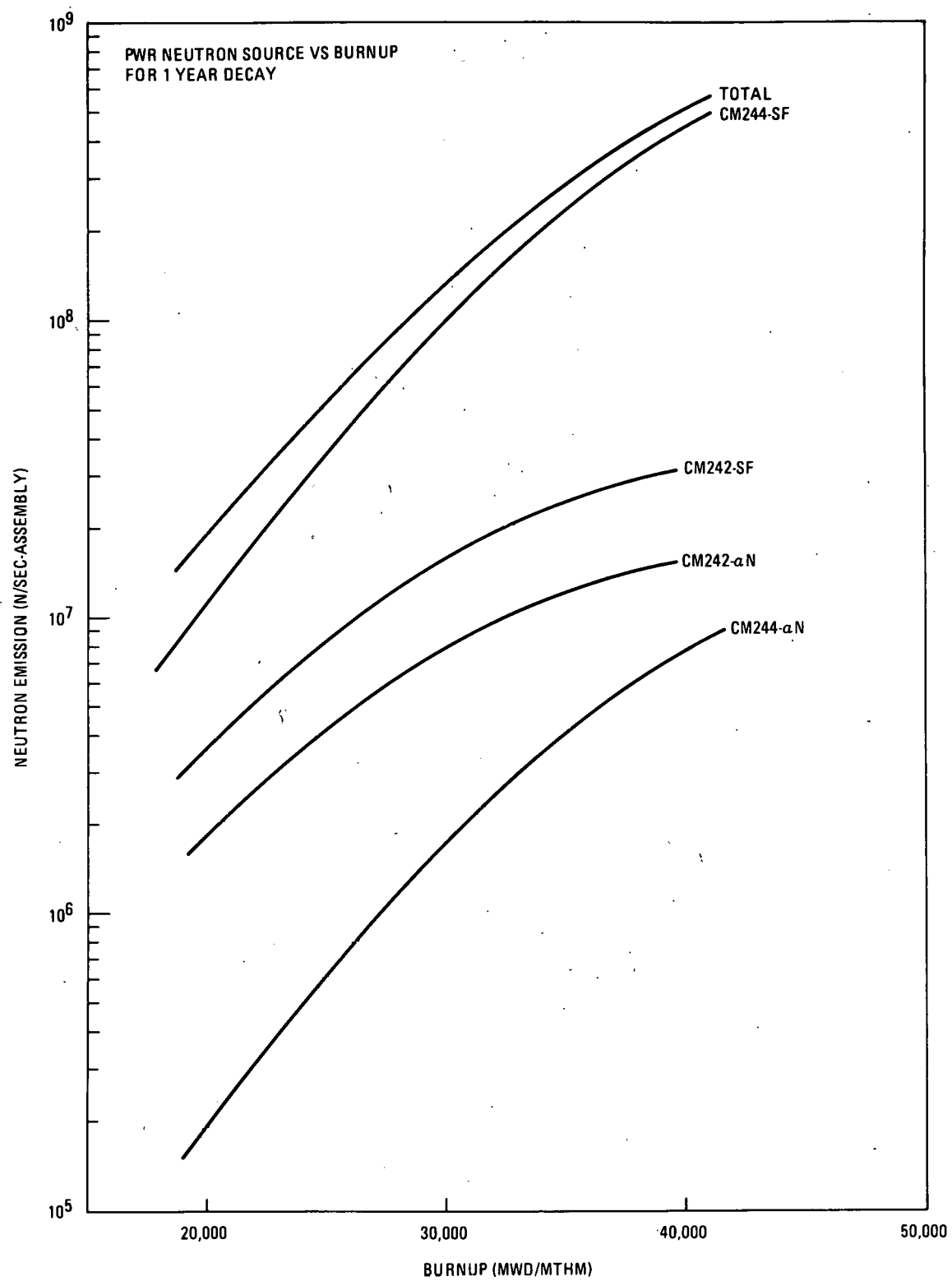
Fig. 3-6. PWR neutron source vs burnup for 1 year decay
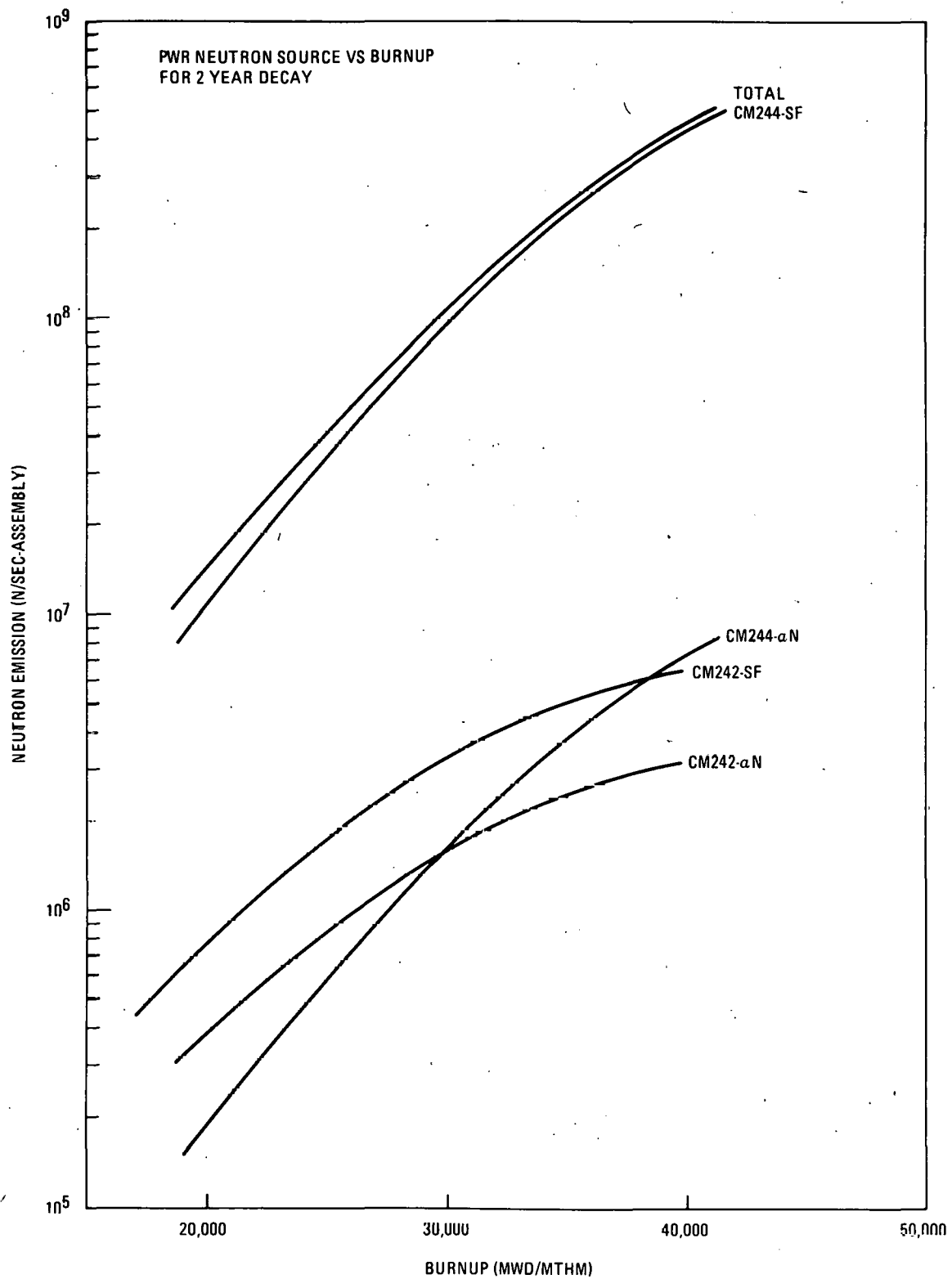
Fig. 3-7. PWR neutron source vs burnup for 2 year decay

38
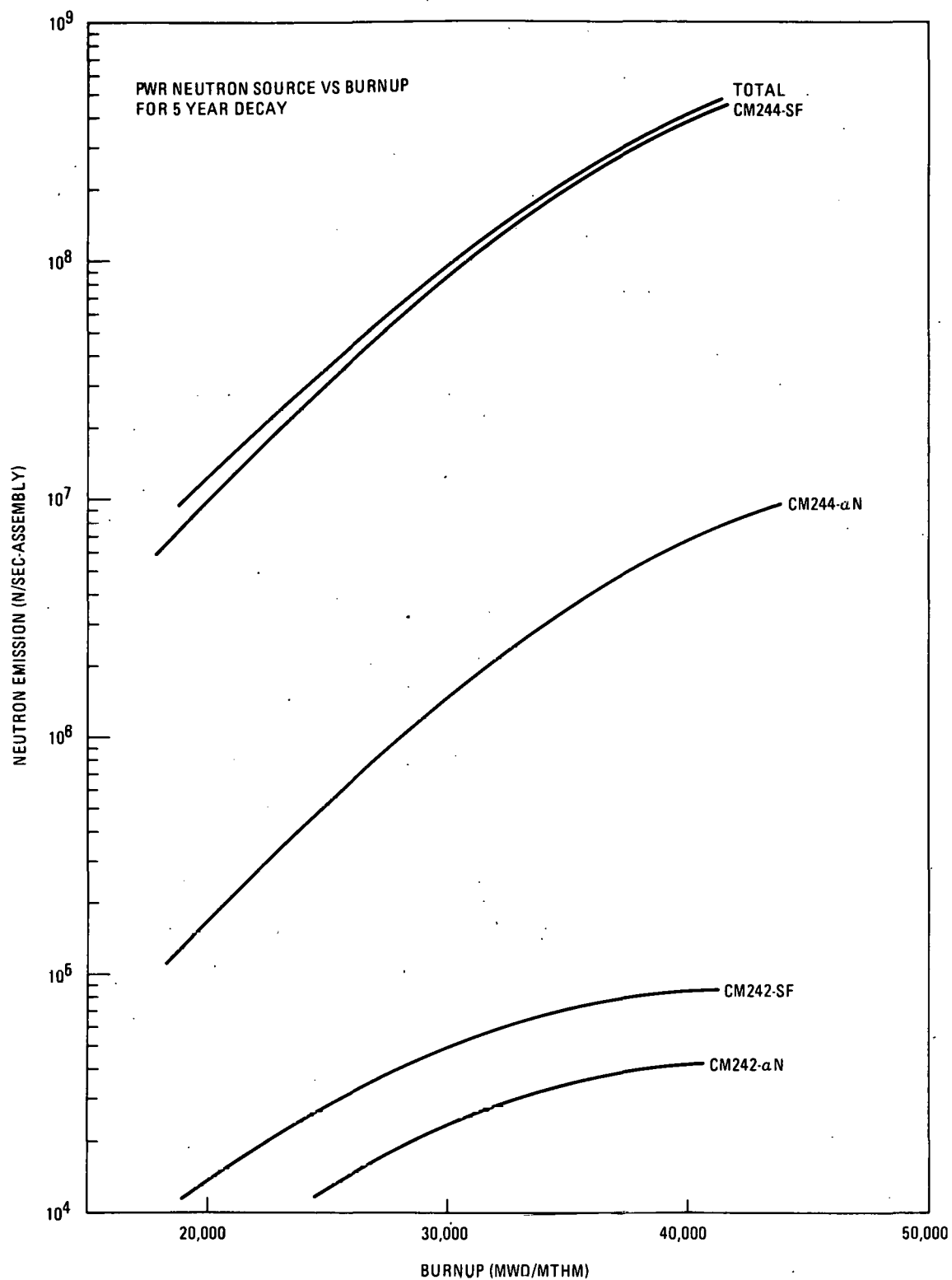
Fig. 3-8. PWR neutron source vs burnup for 5 year decay
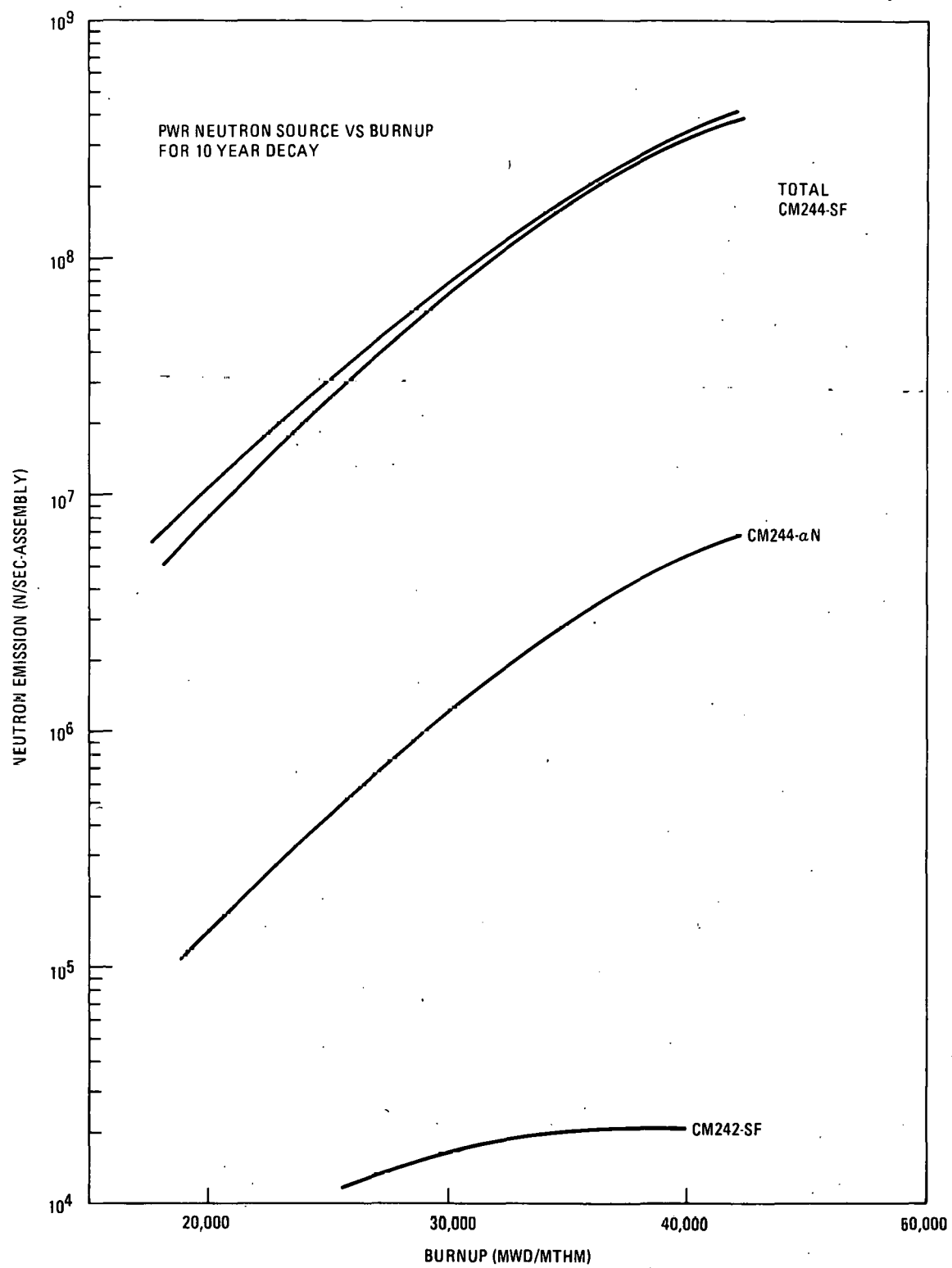
Fig. 3-9. PWR neutron source vs burnup for 10 year decay

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TABLE 3-4
NEUTRON SOURCE (α,n) + SPONTANEOUS FISSION

Fuel	Burnup MWD/MTHM	Decay Time (n/cm^3-sec)			
		1 yr	2 yr	5 yr	10 yr
PWR	20,000	1.022+2	7.715+1	6.548+1	5.737+1
PWR	33,000	1.132+3	9.651+2	8.344+2	6.974+2
PWR	40,000	2.778+3	2.489+3	2.179+3	1.812+3
BWR	34,000	1.718+3	1.487+3	1.29+3	1.077+3
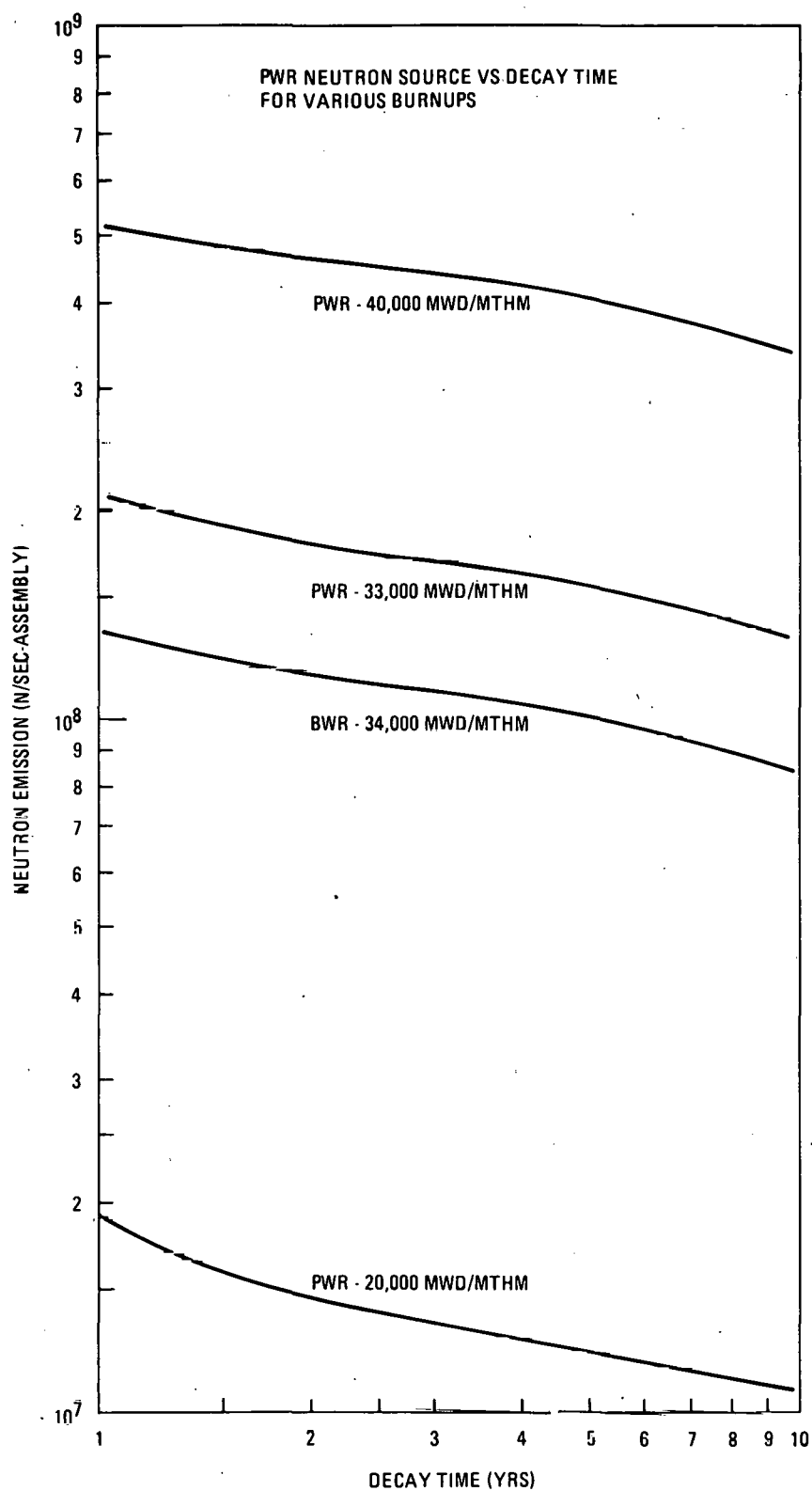
Fig. 3-10. Neutron source vs decay time for various burnups

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TABLE 3-5
NEUTRON SOURCE INPUT INTO DTFX

Energy Range (MeV)	n/cm^3-sec
6.7-14.07	0.014
3.68-6.7	0.106
2.02-3.68	0.417
1.35-2.02	0.157
0.907-1.35	0.108
0.498-0.907	0.117
0.334-0.498	0.080
Total	1.00
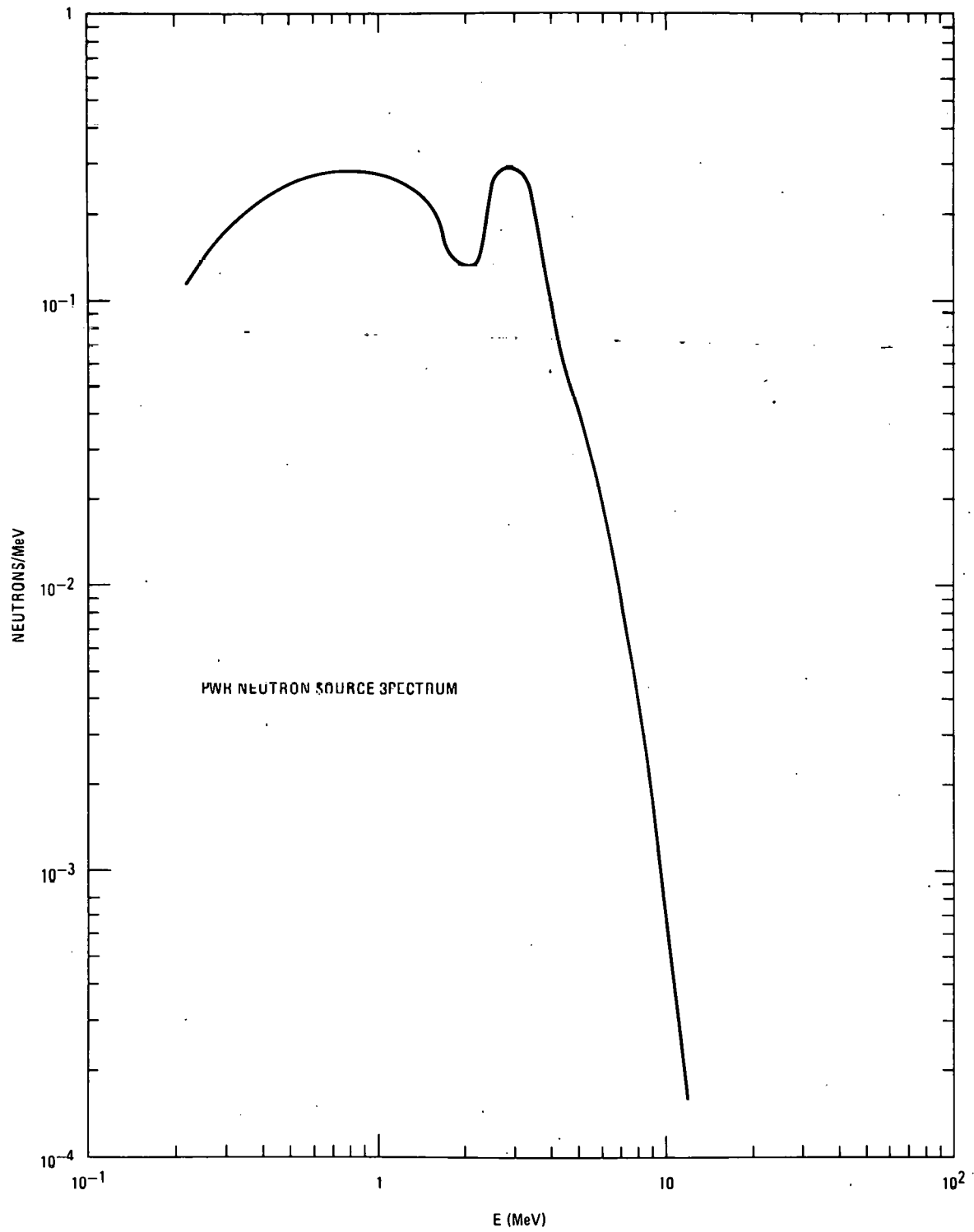
Fig. 3-11.   PWR neutron source spectrum

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The gamma sources used as input for PATH are a combination of source strengths due to actinides,fission products, and activation products. Table 3-6 shows the source strengths broken into six energy groups for PWR and BWR fuel elements with various burnups and decay times. Actually, the only six energy groups input into PATH are those for the case of a fuel element with a specific burnup of 20,000 MWD/MTHM and a decay time of 1 yr. All other dose rate calculations are found utilizing the method in section 3.2.3. It should be noted that the gamma source terms assume no initial cobalt content in the stainless steel structure of the fuel elements. This neglected concentration of about 2000 ppm may change the dose rate results on the top and bottom of the cask where the majority of the fuel assembly structure is located.

TABLE 3-6

GAMMA SOURCE TERMS FOR PATH INPUT $\frac{MeV}{cc/sec}$

$\overline{E}$ (MeV)	DECAY TIME (YEARS)			
	1	2	5	10
PWR 20,000 MWD/MTHM				
.63	1.894 + 10	1.112 + 10	5.46 + 9	3.018 + 9
1.1	9.0 + 8	6.622 + 8	3.584 + 8	1.998 + 8
1.55	5.348 + 8	3.262 + 8	9.485 + 7	1.642 + 7
1.99	5.134 + 8	2.216 + 8	1.529 + 7	2.045 + 5
2.38	1./65 + 7	8.888 + 6	1.135 + 6	3.674 + 4
2.75	1.067 + 8	3.954 + 7	2.085 + 6	3.00 + 4
PWR 33,000 MWD/MTHM				
.63	4.101 + 10	2.513 + 10	1.149 + 10	5.487 + 9
1.1	2.217 + 9	1.702 + 9	9.88 + 8	5.646 + 8
1.55	1.184 + 9	7.657 + 8	2.439 + 8	4.348 + 7
1.99	7.043 + 8	2.926 + 8	2.13 + 7	2.96 + 5
2.38	3.195 + 7	1.609 + 7	2.054 + 6	6.664 + 4
2.75	3.603 + 8	1.328 + 8	6.782 + 6	7.577 + 4
PWR 40,000 MWD/MTHM				
.63	5.652 + 10	3.525 + 10	1.567 + 10	7.022 + 9
1.1	3.099 + 9	2.388 + 9	1.389 + 9	7.907 + 8
1.55	1.678 + 9	1.098 + 9	3.564 + 8	6.393 + 7
1.99	8.654 + 8	3.597 + 8	2.627 + 7	3.697 + 5
2.38	4.211 + 7	2.12 + 7	2.707 + 6	8.808 + 4
2.75	5.828 + 8	2.146 + 8	1.088 + 7	1.124 + 5
BWR 34,000 MWD/MTHM				
.63	3.542 + 10	2.284 + 10	1.06 + 10	5.132 + 9
1.1	2.095 + 9	1.637 + 9	9.811 + 8	5.705 + 8
1.55	1.028 + 9	6.758 + 8	2.205 + 8	3.958 + 7
1.99	4.919 + 8	2.048 + 8	1.505 + 7	2.159 + 5
2.38	2.683 + 7	1.35 + 7	1.725 + 6	5.609 + 4
2.75	3.396 + 8	1.25 + 8	6.361 + 6	7.333 + 4

3.4 COMPOSITION AND GEOMETRY

3.4.1 Compositions

3.4.1.1 Fuel

Fuel assembly compositions vary depending on the reference from which they are obtained. One major factor for this difference is the discrepancy in the fuel assembly structure used in the calculations. One reference may include the end structure while others may not. Table 3-7 shows a comparison between other referenced PWR fuel elemental compositions and the conservative averages used in this study.

To determine conservative averages for BWR fuel elemental compositions a comparison was made between PWR and BWR taken from Ref. 3-18 and shown in Table 3-8. Note that when the elemental compositions are compared for PWR and BWR fuel assemblies in units of grams per unit cross sectional area the values obtained are very close to one another. With this data the conclusion is that the same volumetric atomic concentrations as those found in PWR fuel assemblies may be used for BWR fuel assemblies. The actual nuclide concentrations used for the fuel region in this study are found in Table 3-9.

From Table 3-9 a change was made since Zr cross sections were not available in the General Atomic data base. For DTFX calculations, Mo cross sections were adjusted to resemble as close as possible, the Zr cross sections. To accomplish this, the Mo absorption cross section was reduced to zero and the total cross section was reduced accordingly. Fig. 3-26 shows the relation between the actual Zr cross section and that used as input into DTFX. For PATH calculations Mo gamma attenuation data was substituted for that of Zr.

3.4.1.2 Neutron Shield Material

The neutron shield material chosen for this study is a commercial product named Boro-Silicone, with properties as specified in Ref. 3-18. This material has a recommended temperature limit of $400^\circ F$ ($205^\circ C$) along with properties of good machinability and capability of casting in the field. Table 3-10 lists the nuclides and their weight percent along with their atomic densities.

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TABLE 3-7
PWR COMPOSITION COMPARISON

Element	Grams per Assembly		
	National Lead (Ref. 3-12)	Sandia (Ref. 3-17)	Conservative Averages Used in This Study
O	57,300	61,400	60,000
Cr	122	3,110	
Fe	471	8,105	
Ni	2,530	3,825	
Zr	102,000	125,800	100,000
U-238	396,000	427,000	400,000
U-235	2,500	2,400-3,500	3,000
Pu-239	1,990	2,425	2,500
Pu-240	548	1,000-1,100	1,000
Si		100	
Mn		200	
Nb		280	
Mo		170	
Sn		2,054	

Note: Sandia's composition includes the end structure of the fuel assembly. For conservatism, these structural elements were omitted for side dose rate calculations in the present study.

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TABLE 3-8
BWR COMPOSITION COMPARISON BASED ON SAND79-0172

Element	PWR Grams per Assembly	BWR Grams per 3 Assemblies	PWR Grams per Unit Cross Sectional Area	BWR Grams per Unit Cross Sectional Area
Zr	125,800	161,760	244	252
O	61,400	73,800	119	115
U-238 U-235	456,000	547,000	885	851
Pu-239	2,430	3,000	4.72	4.67
Pu-240	1,000-1,100	1,280	2	2

Conclusion: Use the same volumetric atom concentrations for BWR as for PWR.

TABLE 3-9
NUCLIDE CONCENTRATIONS FOR THE FUEL

Nuclide	gm/assembly	at/b-cm
O	6.0+4	1.2-2
Zr	1.0+5	3.51-3
U-238	4.0+5	5.39-3
U-235	3.0+3	4.09-5
Pu-239	2.5+3	3.41-5
Pu-240	1.0+3	1.34-5
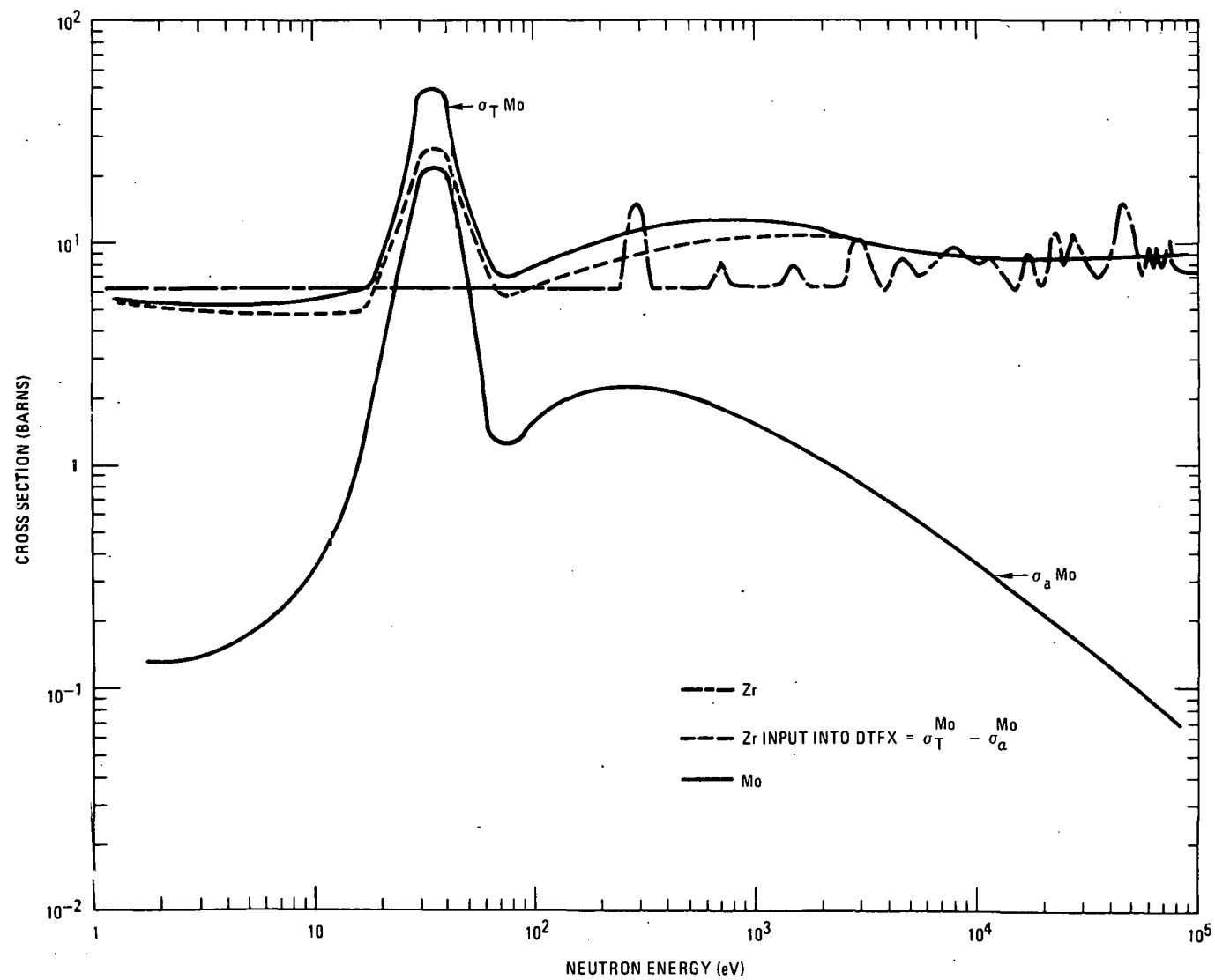
Fig. 3-26. Zr and Mo neutron cross section vs. neutron energy

TABLE 3-10
NUCLIDE CONCENTRATIONS FOR BORO-SILICATE
(NEUTRON SHIELD)

Nuclide	wt %	at/b-cm
O	46.77	2.82-2
Si	17.63	6.05-3
Al	17.36	6.20-3
C	10.9	8.75-3
H	4.63	4.43-2
Ca	1.03	2.48-4
B-10	0.20	1.8-4
B-11	0.81	7.21-4
Na	0.39	1.64-4
Mg	0.19	7.54-5

3.4.1.3 Gamma Shield Material

Depleted uranium was taken as 0.2 wt % U-235 and 99.8 wt % U-238. These nuclide concentrations along with those for air and iron are shown in Table 3-11.

3.4.2 Geometry

3.4.2.1 Side

The geometry for a Westinghouse 17 x 17 PWR fuel assembly was obtained from Ref. 3-16 and is shown in Fig. 3-12. For all PWR calculation, one fuel assembly was considered placed inside a 12-in. schedule 40 pipe 12 ft. long and centered inside the cask. For side calculations performed by PATH the fuel was taken as rectangularly homogenized and positioned in the geometry shown in Fig. 3-13. Dose points were located off the corner of the fuel element where the highest dose rates would be obtained. For DTFX side calculations, the cylindrical geometry shown in Fig. 3-14 was utilized. In this case the iron pipe and steel structure inside the cask have been homogenized to fill the area between the fuel and depleted uranium regions. The equivalent thickness of iron is 2.6 cm. in this region.

The geometry for a General Electric 8 x 8 BWR fuel assembly was also obtained from Ref. 3-16 and is shown in Fig. 3-15. For all BWR dose rate calculations, three homogenized fuel elements were considered placed inside a 16 inch schedule 40 pipe and centered inside the cask. The PATH geometry is shown in Fig. 3-16 along with the dose point locations. The geometry for BWR side calculations performed by DTFX (Fig. 3-17) is the same as that for the PWR case except that the reduced density iron between the fuel and depleted uranium has an equivalent thickness of 2.9 cm.

3.4.2.2 Top

Due to the nature of the obtainable information of PWR and BWR fuel assemblies, the top and bottom configurations were taken as the same for both. The top of the fuel assembly was broken down into the 5 regions shown in Fig. 3-18. All regions are homogenized except for the upper tie plate region in which the actual geometry was used.

TABLE 3-11

NUCLIDE CONCENTRATIONS FOR GAMMA
SHIELDING MATERIALS

Material	Nuclide	at/b-cm
Depleted Uranium	U-235	9.69-5
	U-238	4.77-2
Air	N	4.9-5
Iron	Fe	8.51-2

0.875 DIA REF

7.162 REF

8.404 SQ REF

TOP VIEW

0.875 DIA REF

6.750 REF

8.424 SQ REF

BOTTOM VIEW

CONTROL ROD

ROD CLUSTER CONTROL ASSEMBLY

FUEL ROD

GRID ASSEMBLY

151.63 REF

BOTTOM NOZZLE

TOP NOZZLE

2.7 REF

159.8 REF

55
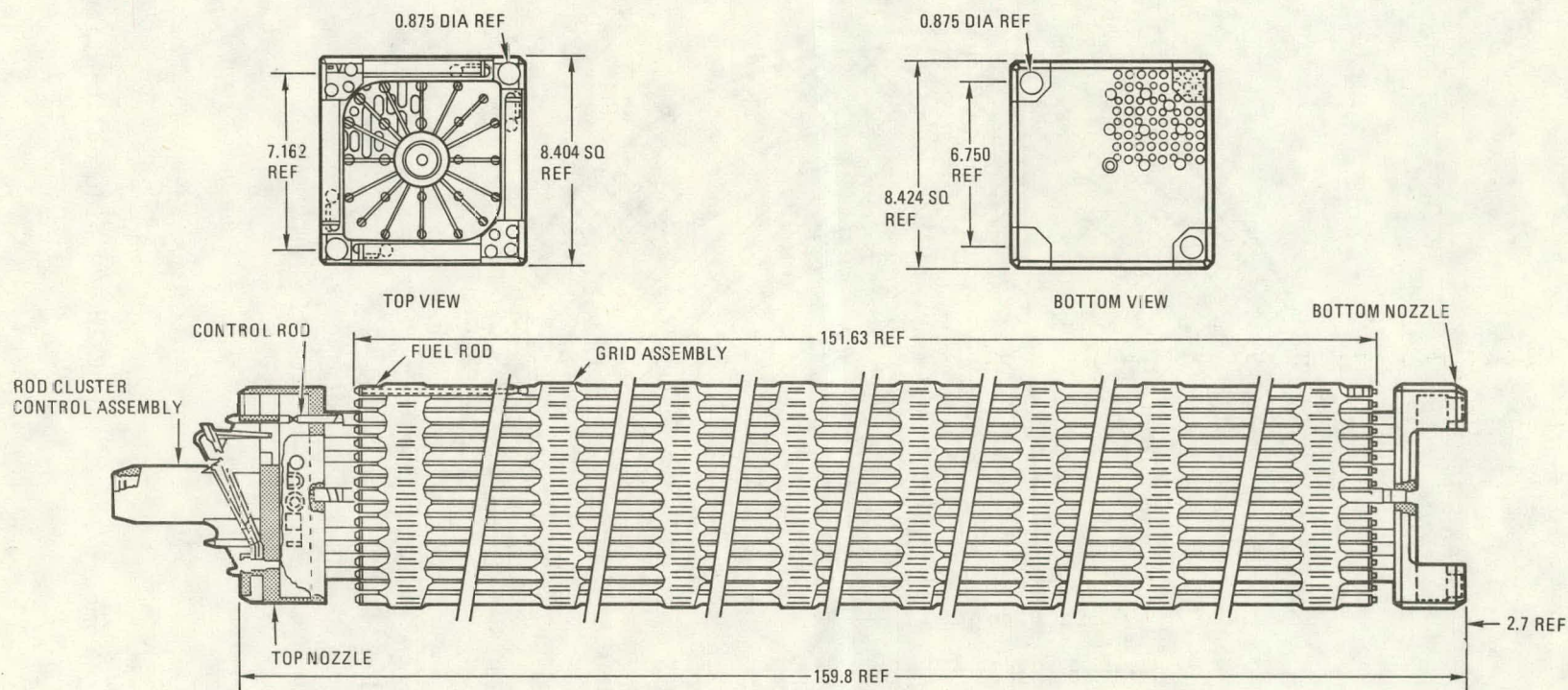
Fig. 3-12. Westinghouse 17 x 17 PWR Fuel Assembly
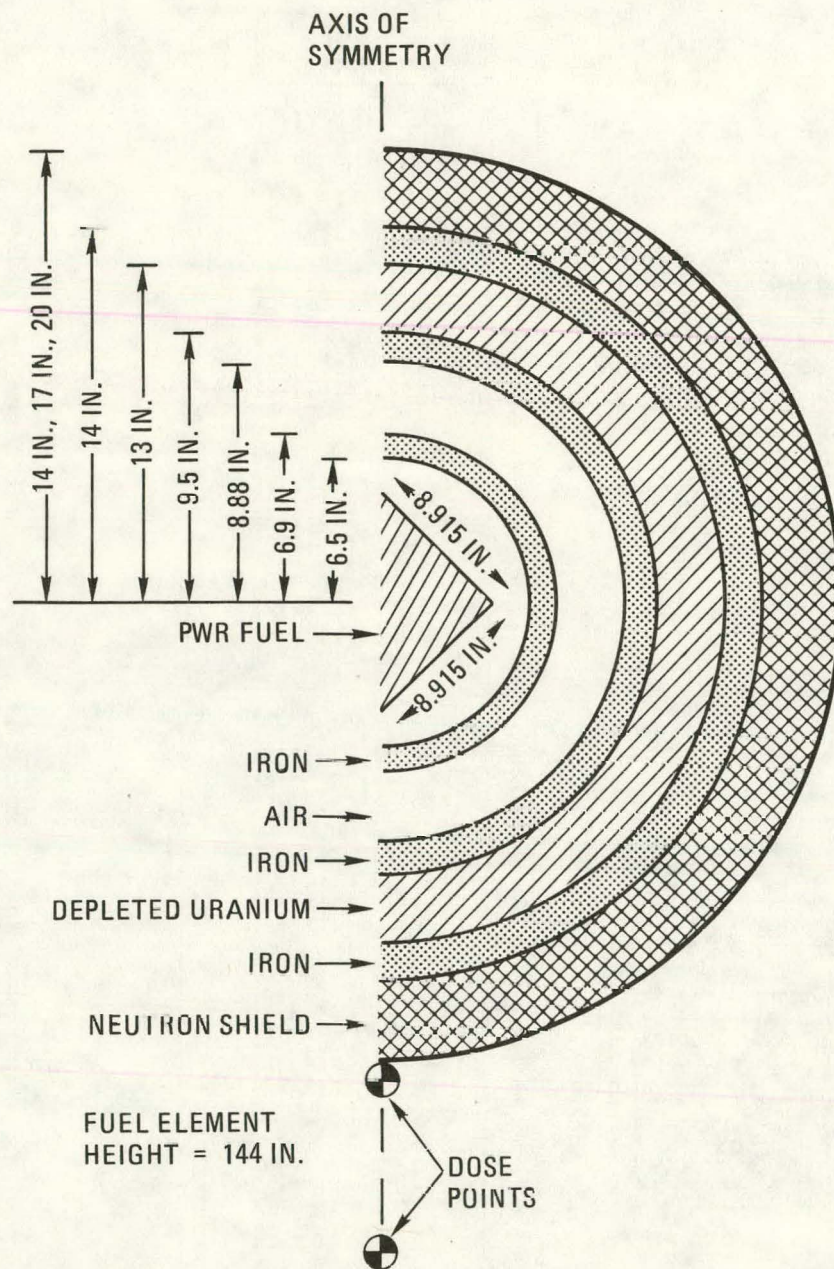
Fig. 3-13. Geometry input for **PATH** PWR side calculation

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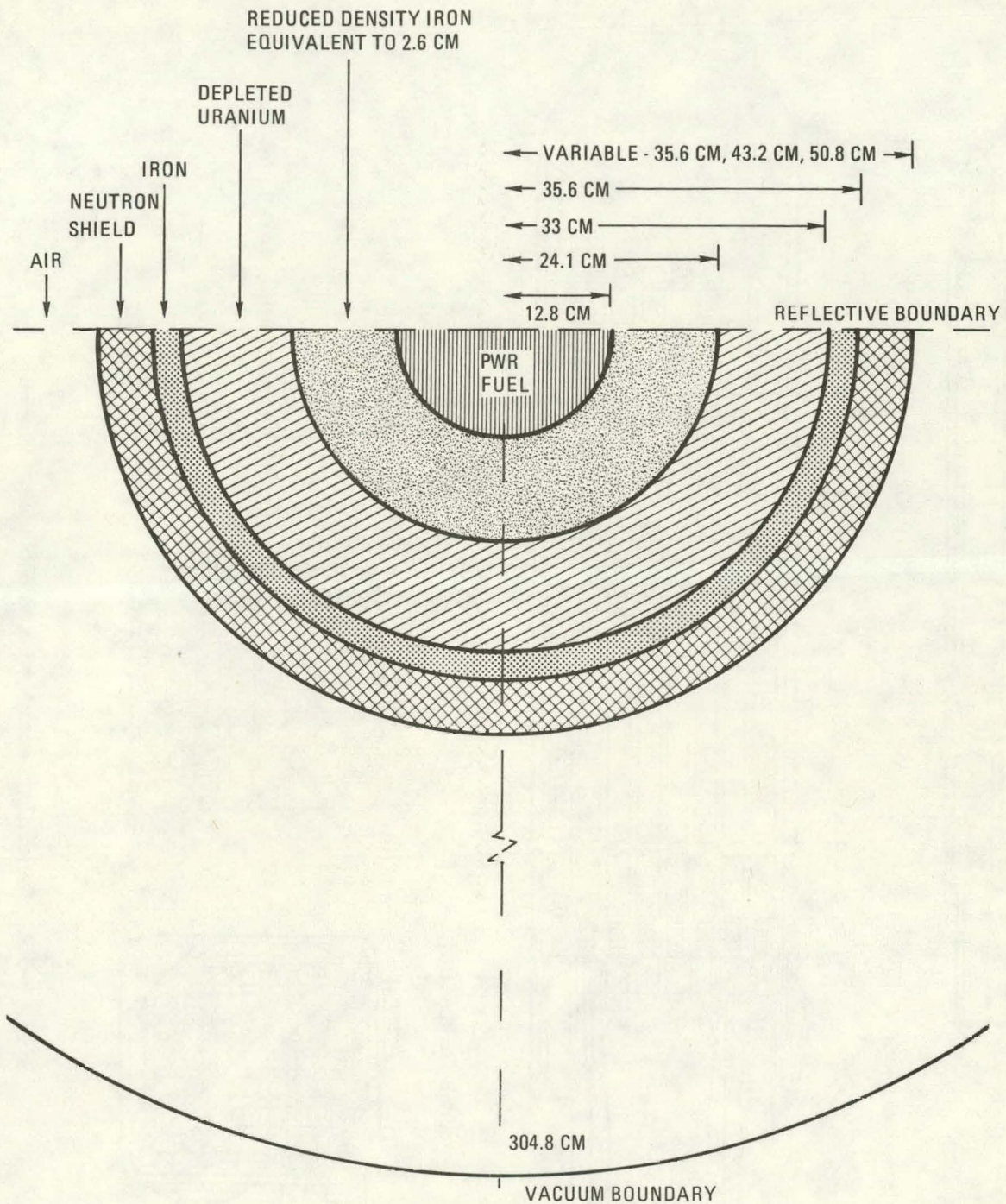
Fig. 3-14.   Geometry input for DTFX-PWR side calculation

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TYPICAL FUEL ROD

ACTIVE FUEL LENGTH

UPPER TIE PLATE

CHANNEL

SPACER,
TYPICAL
OF SEVEN

LOWER
TIE PLATE
CASTING

SPACER DETAIL

W — SPACER POSITIONING/WATER ROD
T — TIE ROD
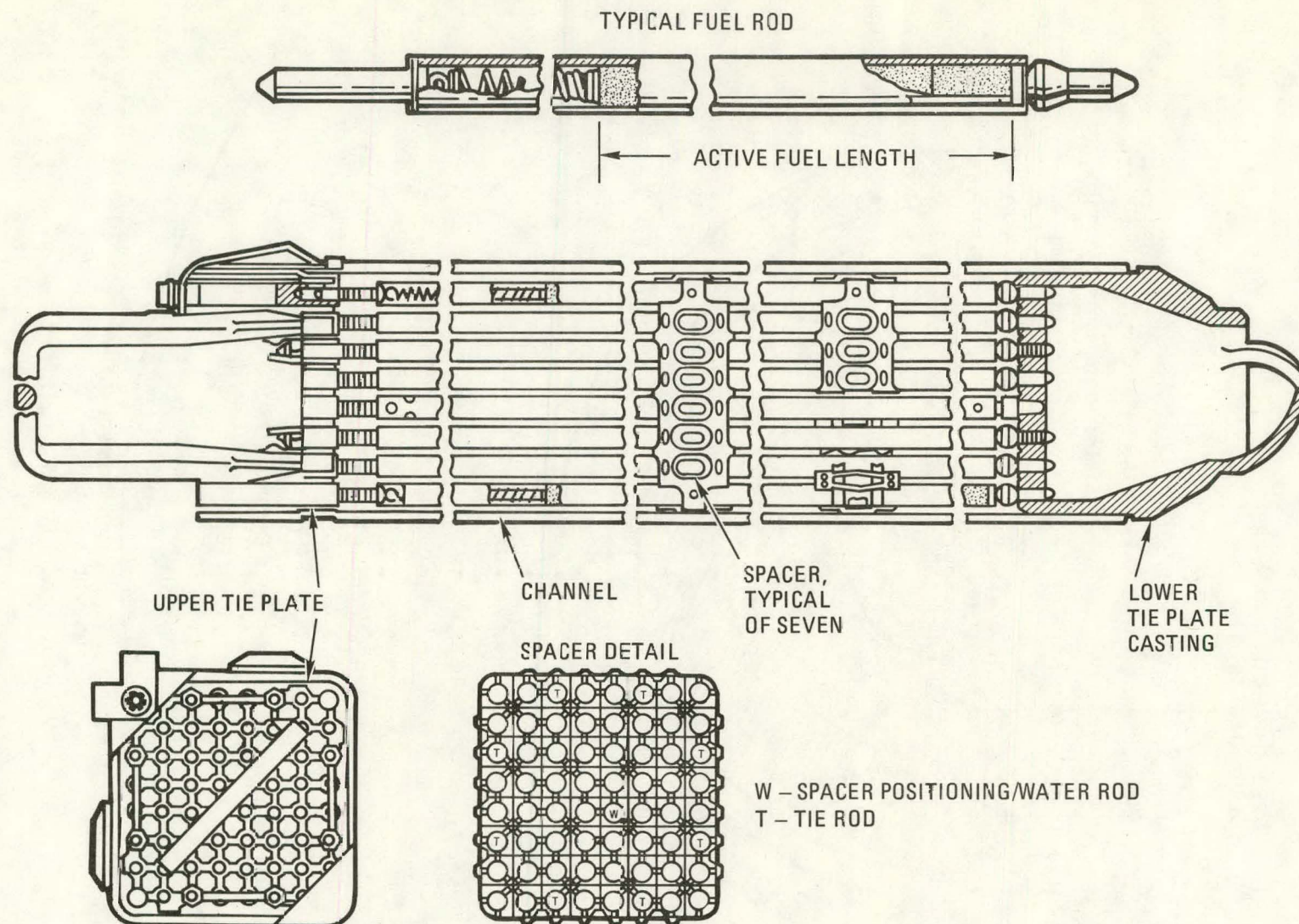
Fig. 3-15.   General Electric 8 x 8 BWR Fuel Assembly
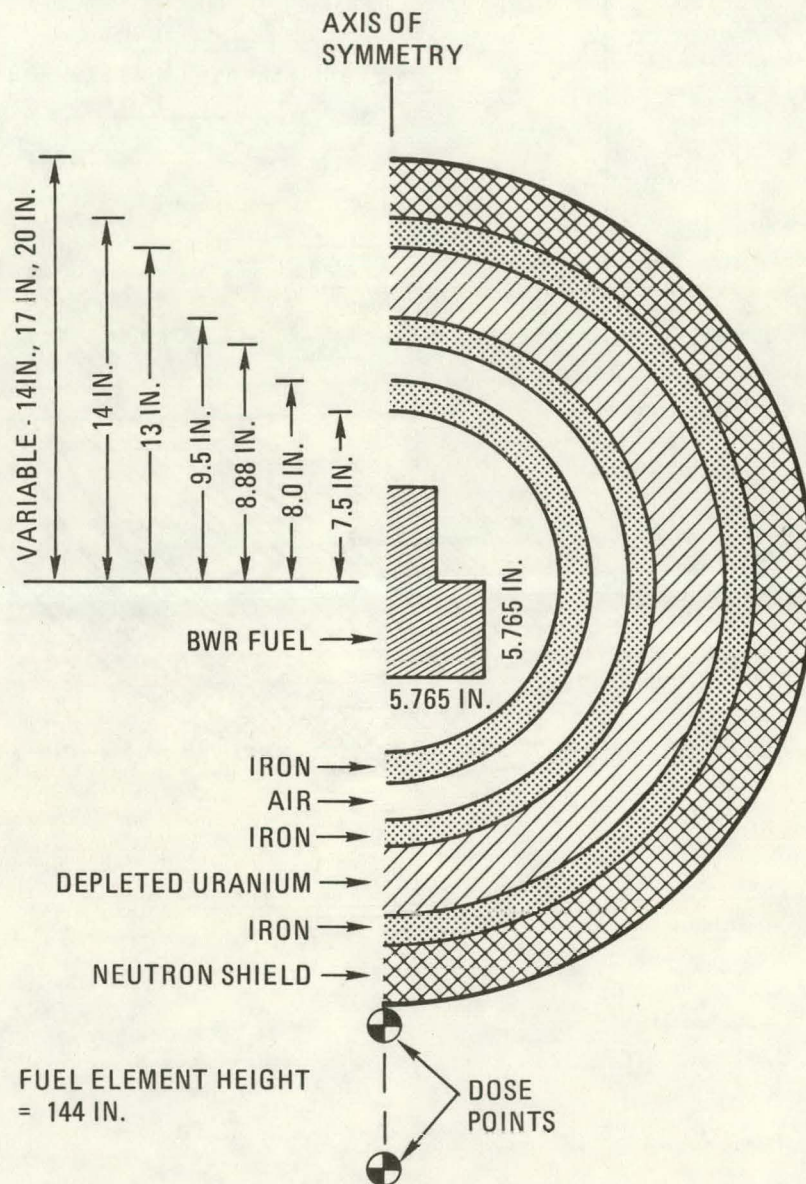
Fig. 3-16. Geometry input for **PATH** BWR side calculation
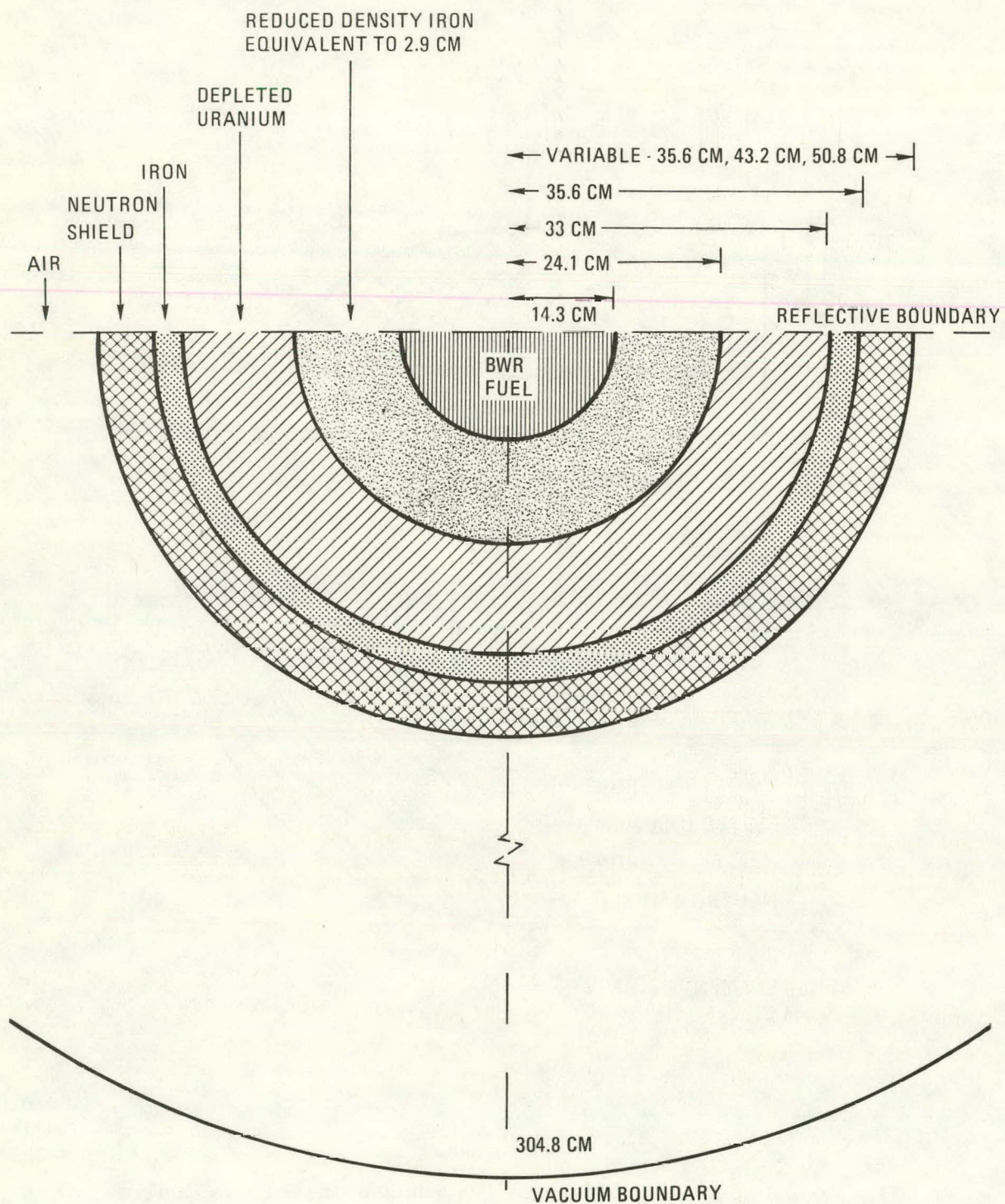
Fig. 3-17. Geometry input for DTFX-BWR side calculation

60
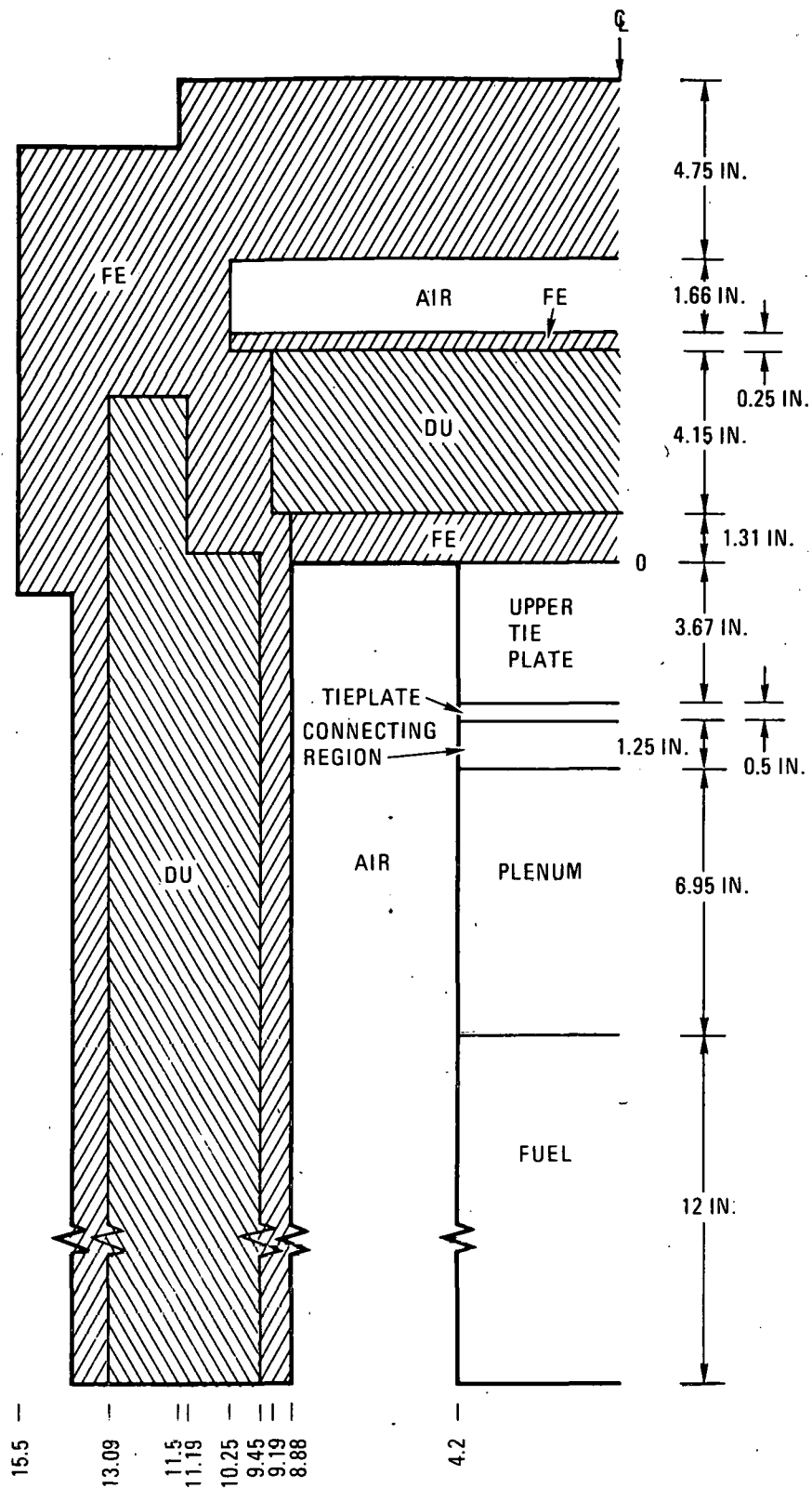
Fig. 3-18. Geometry input for PATH top of FSV-1 cask

The homogenized densities of each region were calculated from actual dimensions found in Ref. 3-16. These fuel regions were placed as close to the top of the cask as possible, so that conservative dose rates could be obtained. Two designs of cask closures were considered. Both cases consider the closures (Fig. 3-19a) with the inner container of the cask removed. In the first case the inner closure is lower in the cask body by the thickness of the container flange and has an air gap formed above it. The PATH geometry is shown in Fig. 3-18 and all regions are included. The geometry input for DTFX appears in Fig. 3-20. All low density regions have been removed so as to reduce unnecessary computer time. The second case considers an entirely new closure design, as shown in Fig. 3-21. The geometry for PATH calculations is drawn in Fig. 3-22. The geometry input for DTFX is shown in Fig. 3-23. Once again, all low density regions have been removed. In both closure designs the dose rate calculations were performed at dose point locations where the dose rate is a maximum.

3.4.2.3 Bottom

The bottom of the FSV-1 cask is shown in Fig. 3 19b. This figure also includes the 2-in. steel bottom of the inner container. In considering the shielding analysis previously done on the cask (Ref. 3-4) this extra 2 inches of steel was retained in our analysis even though the inner container has been removed. The PATH geometry for this case is shown in Fig. 3-24 along with the location of the dose points. The lower tie plate region of the fuel assembly has been ignored because its geometry does not come into considerations when dose points are located along the center axis of the fuel assembly. Also, the iron thicknesses of both the fuel rod bottom end plug and grill area of the lower tie plate have been added together and considered as one region. For the geometry of DTFX calculations (Fig. 3-25) all iron in consideration was placed into one region while all low density material inside the cask was removed. In both the DTFX and PATH dose rate calculations the dose point locations extended 6 ft. beyond the edge of the transport vehicle.
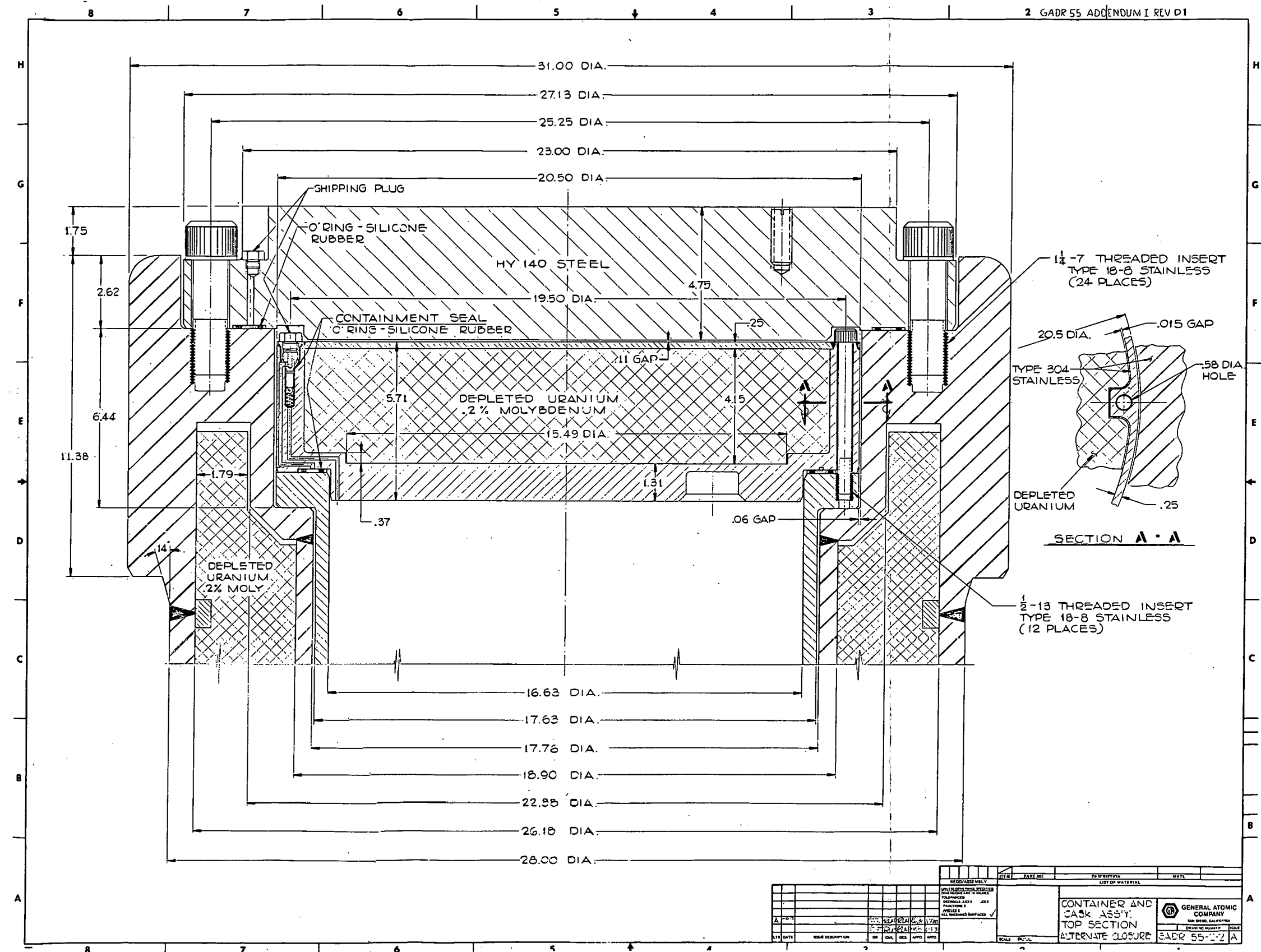
Fig. 3-19a. Top end closure of as-built cask

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Fig. 3-19b. Bottom of FSV-1 Cask
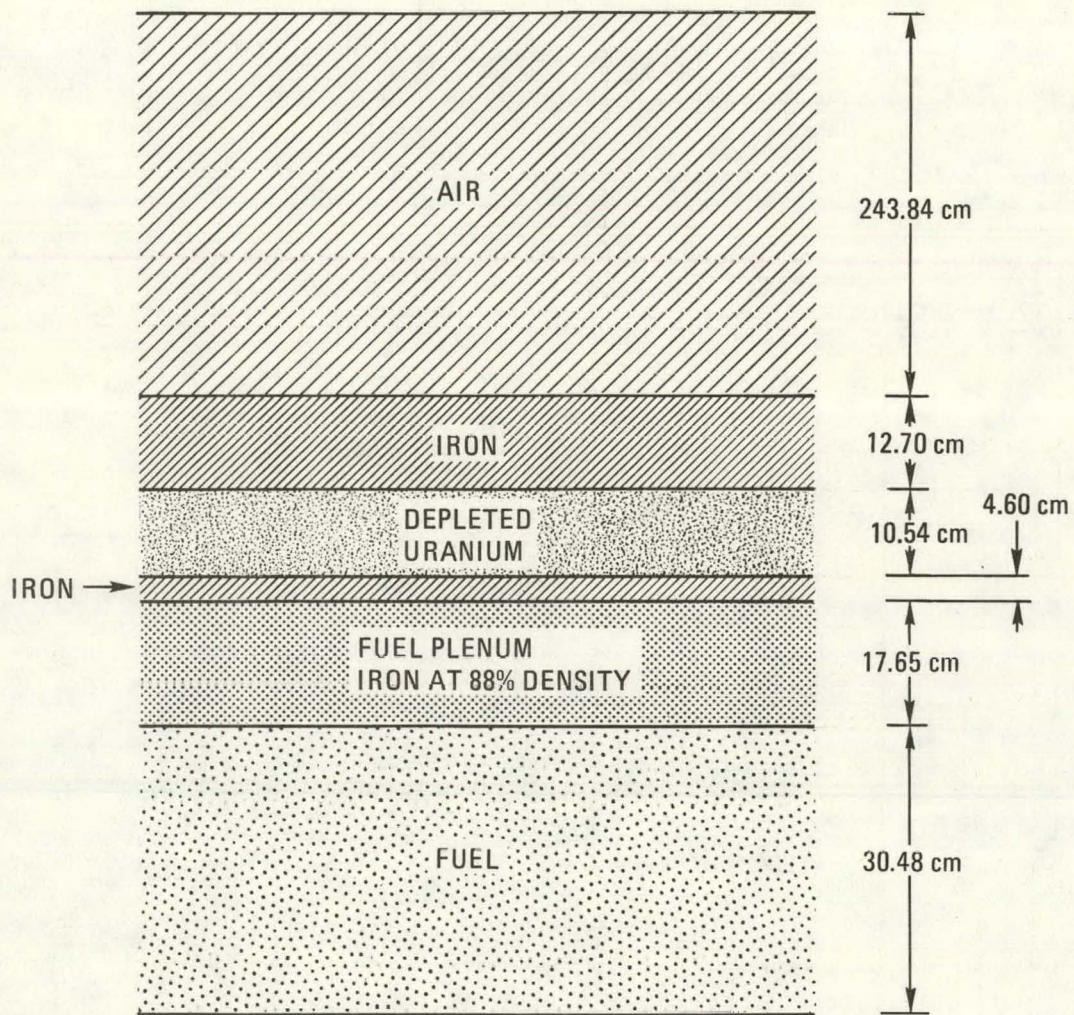
Fig. 3-20. Geometry for top of FSV-1 cask for input into DTFX
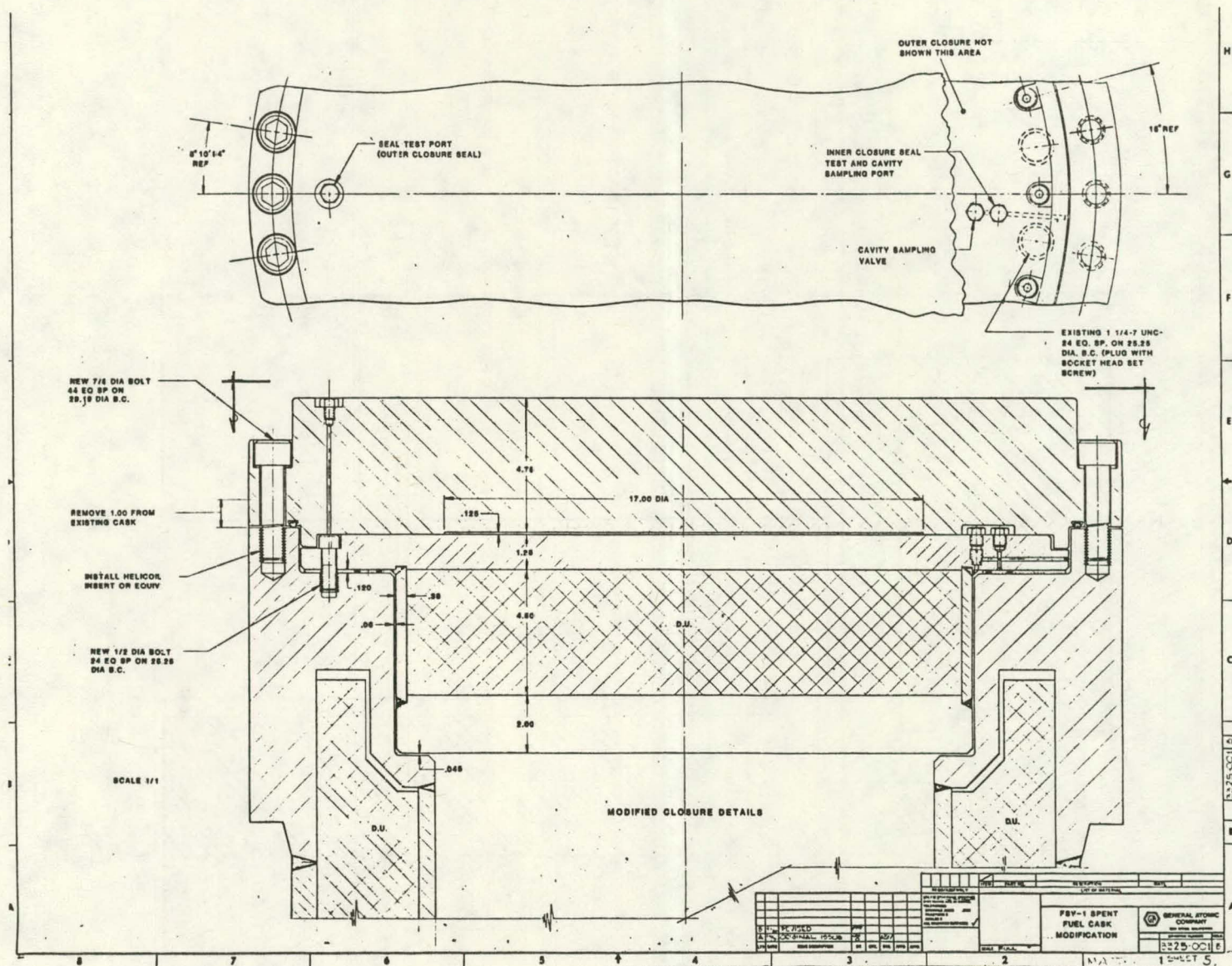
Fig. 3-21. New end closure design
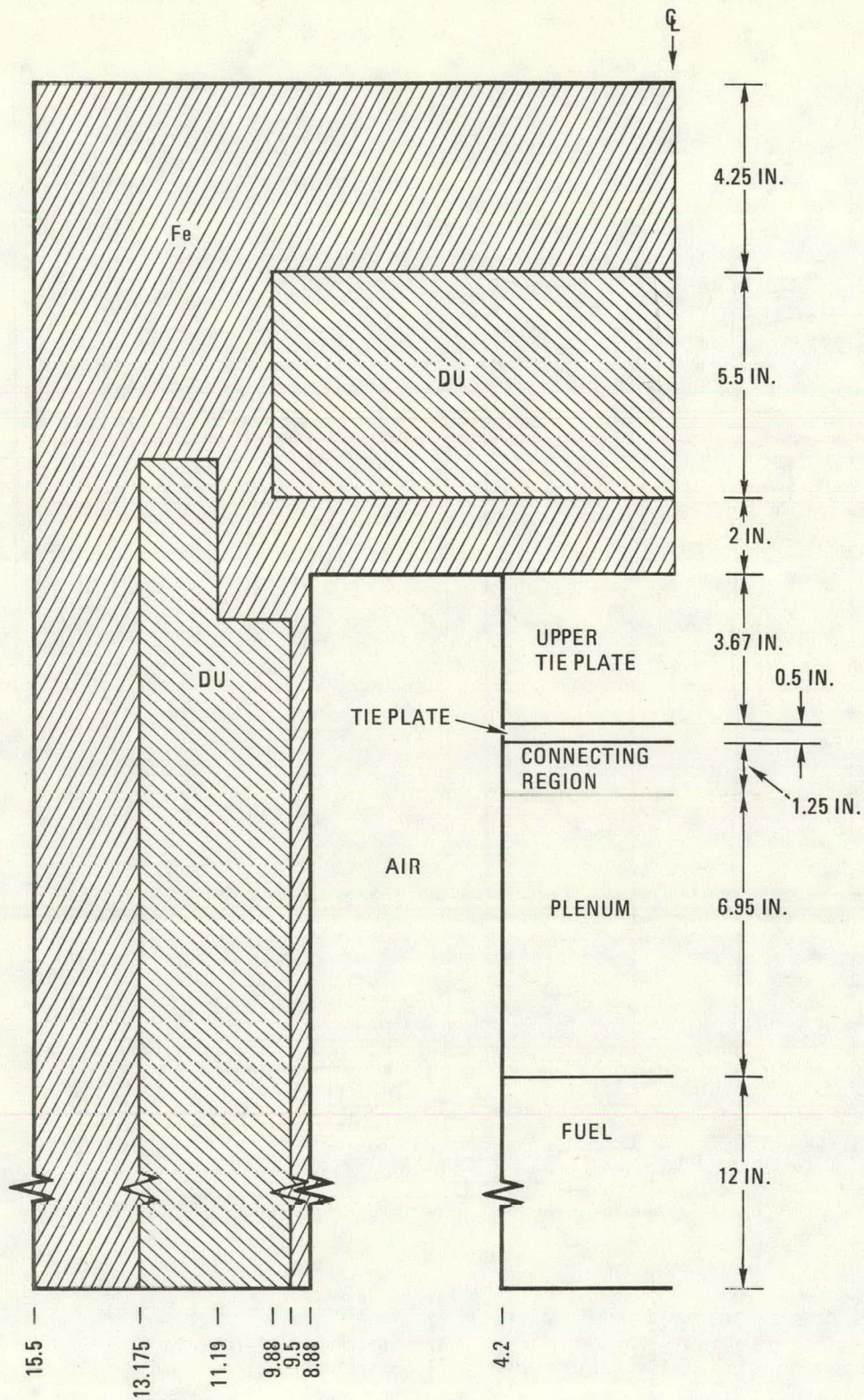
Fig. 3-22. Geometry input for PATH top of FSV-1 cask --
new design

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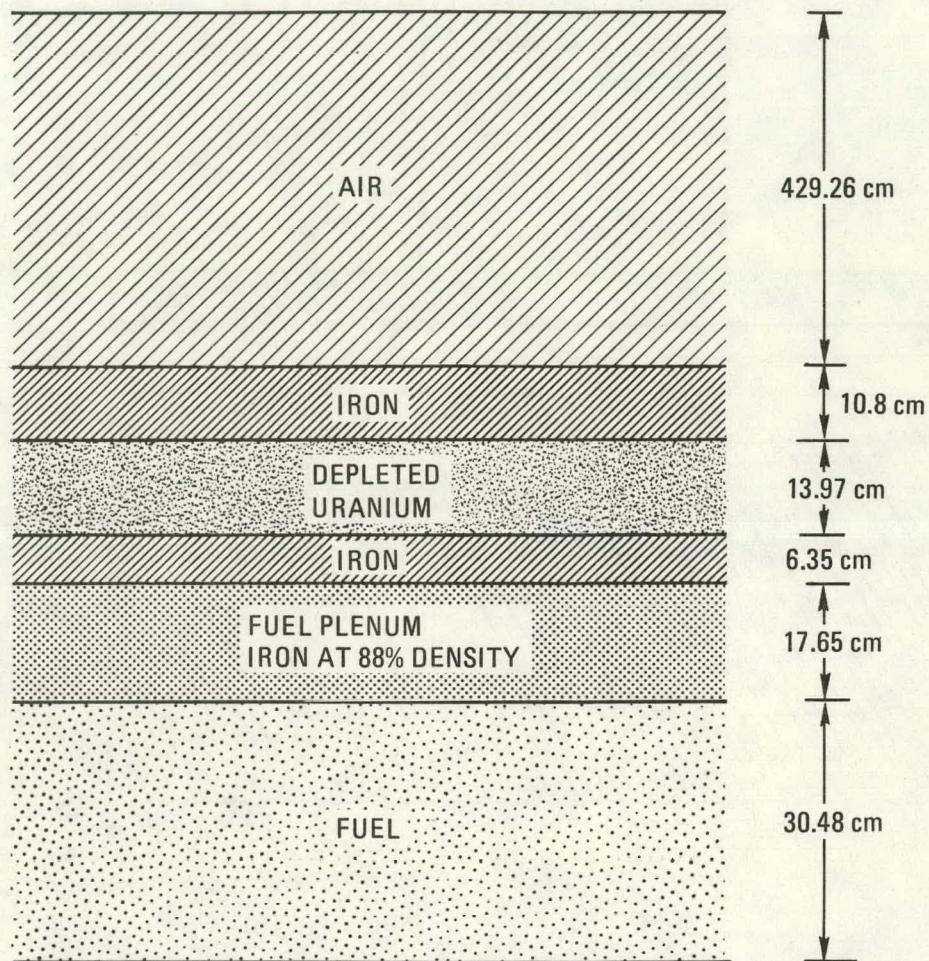
Fig. 3-23. Geometry for input for DTFX top of FSV-1 cask — new design
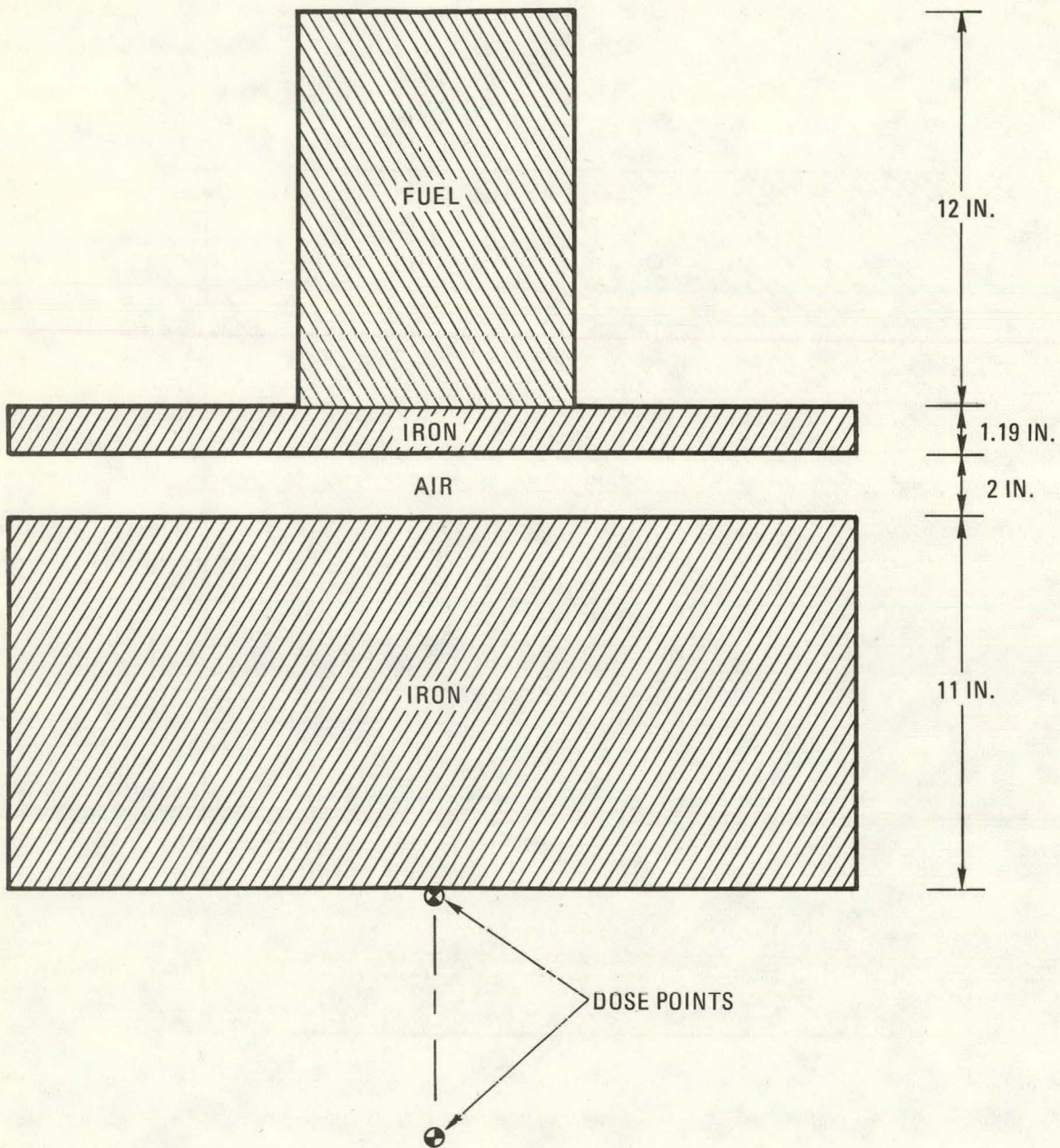
Fig. 3-24. Geometry input for PATH-bottom of FSV-1 cask
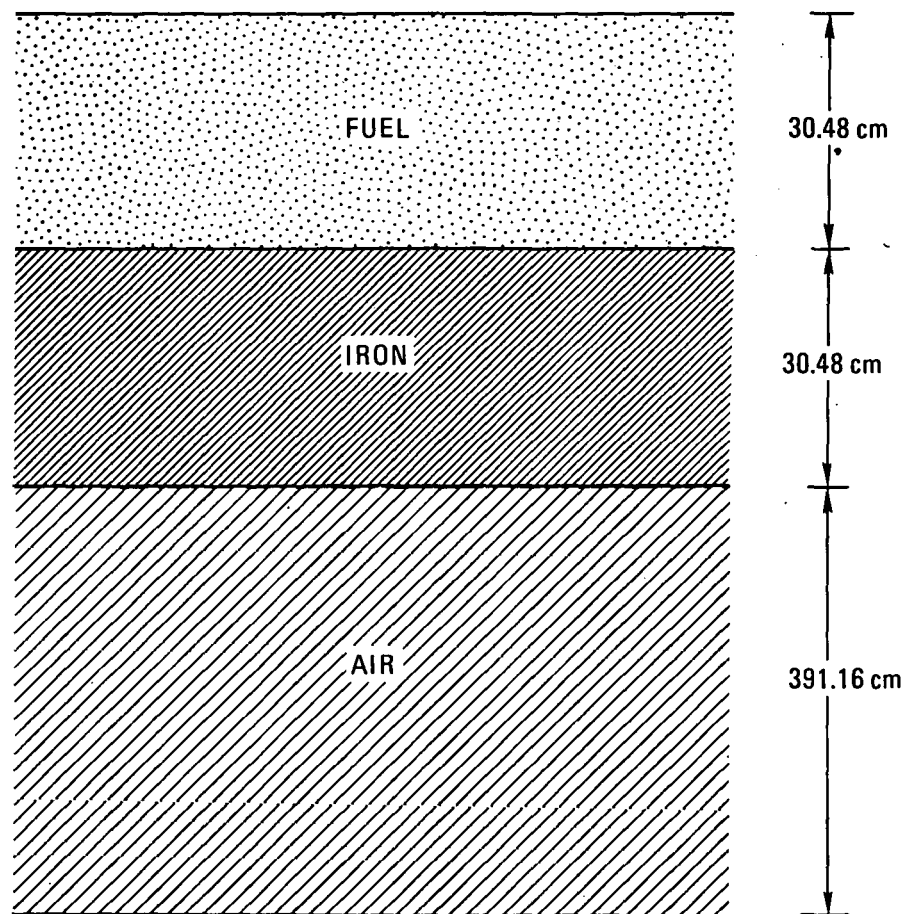
Fig. 3-25. Geometry of input for DTFX – bottom of FSV-1 cask

3.5 RESULTS

3.5.1 PWR

3.5.1.1 Side

Dose rates from the side of the cask were considered for both PWR and BWR fuel assemblies. PATH calculations considered dose rates on the side of the cask and 6 ft. from the edge of the transport vehicle, for neutron shield thicknesses of 0, 3, and 6 in. Dose rate calculations performed by DTFX considered 0 and 3 in. neutron shield for the same dose points as those in PATH.

3.5.1.1.1 Neutrons.

The results of neutron dose rates 6 ft from the edge of the transport vehicle due to one PWR fuel assembly within the cask are shown in Table 3-12 and Figs. 3-27 and 3-28 for various combinations of burnup, decay time, and neutron shield with no added extra gamma shield.

The total neutron dose rate per source neutron has been broken down into primary and secondary contributions in Fig. 3-29 for the different regions in the as-built cask. Also shown is the secondary gamma dose rate due to the primary source neutrons. The neutron dose rate breakdown is shown again in Fig. 3-30 for the case of 3-in. of neutron shield added to the cask. It is interesting to note that secondary neutrons contribute 15% to 35% as much to the dose rate as do the primary neutrons.

Values of neutron dose rates for other shielding combinations may be found with the help of Figs. 3-3 and 3-4 and the method described in Section 3.2.2.

3.5.1.1.2 Gammas.

The results of gamma side dose rates 6 ft from the edge of the transport vehicle are shown in Figs. 3-31 and 3-32 at various decay times and burnups for the case with no extra neutron or gamma shielding. Dose rates for other cases are found by using the data shown in Figs. 3-3 and 3-4 and the methods described in Section 3.2.2.

Dose rate information regarding a change in dose point location may need to be known. Figure 3-33 normalizes the total dose rate to the side of the cask and allows one to transform dose rates to any location between the cask and 6 ft from the edge of the transport vehicle.

TABLE 3-12
NEUTRON DOSE RATES 6 FT FROM EDGE OF TRANSPORT VEHICLE
MREM/HR

Neutron Shield (in.)	Gamma Shield (in.)	Burnup (MWD/MTHM)	Decay Time			
			1 yr	2 yr	5 yr	10 yr
			PWR			
0	0	20,000	1.71	1.29	1.09	0.96
		33,000	18.90	16.10	13.90	11.70
		40,000	46.40	41.60	36.40	30.30
3	0	20,000	0.20	0.15	0.13	0.11
		33,000	2.20	1.88	1.62	1.36
		40,000	5.40	4.84	4.24	3.52
6	0	20,000	0.01	0.01	0.01	0.01
		33,000	0.14	0.12	0.10	0.09
		40,000	0.35	0.31	0.28	0.23
			BWR			
0	0	34,000	34.60	29.90	26.00	21.70
3	0	34,000	3.99	3.45	3.00	2.50
6	0	34,000	0.26	0.22	0.20	0.16
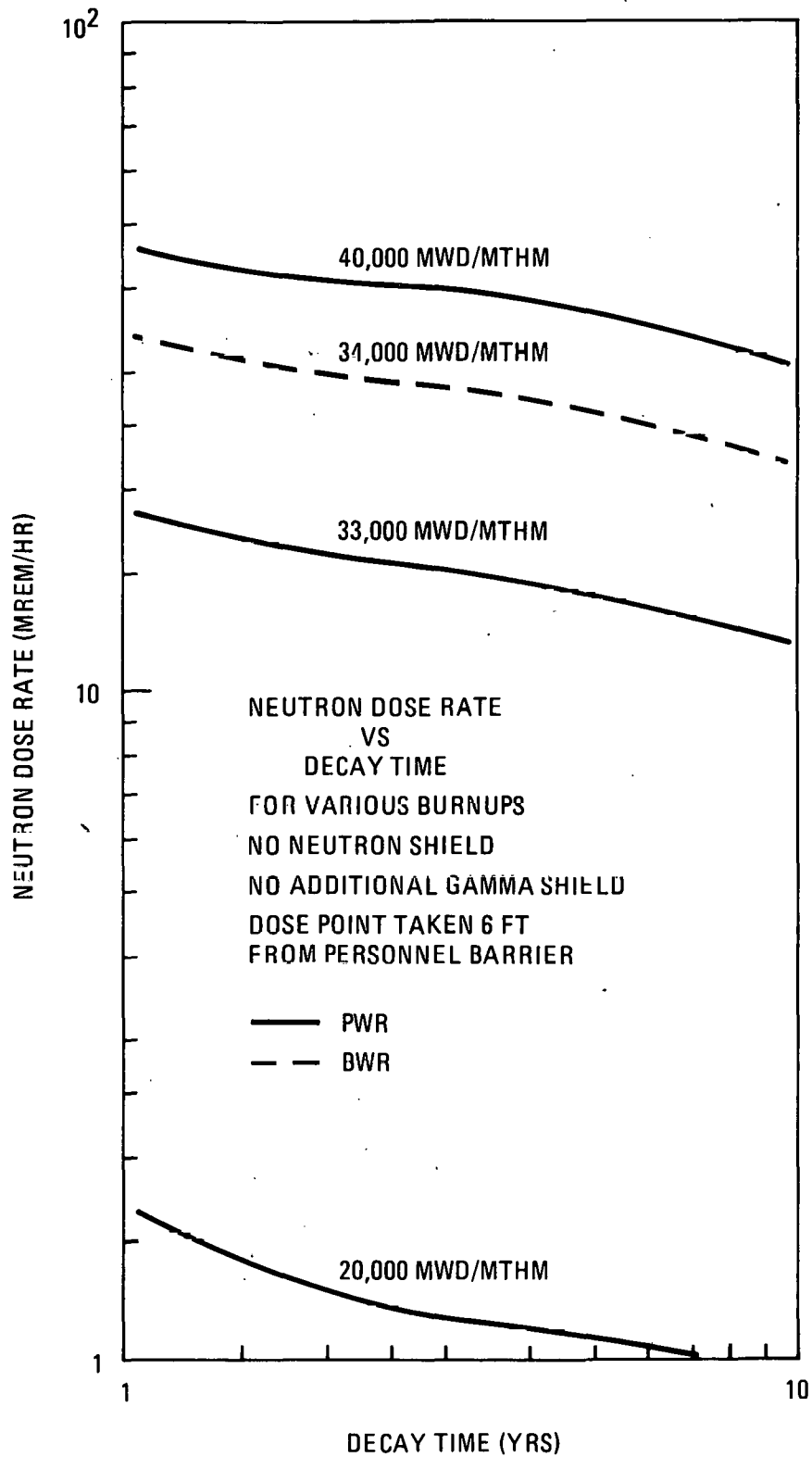
Fig. 3-27. Neutron dose rate vs decay time for various burn-
ups, no neutron shield, no additional gamma
shield
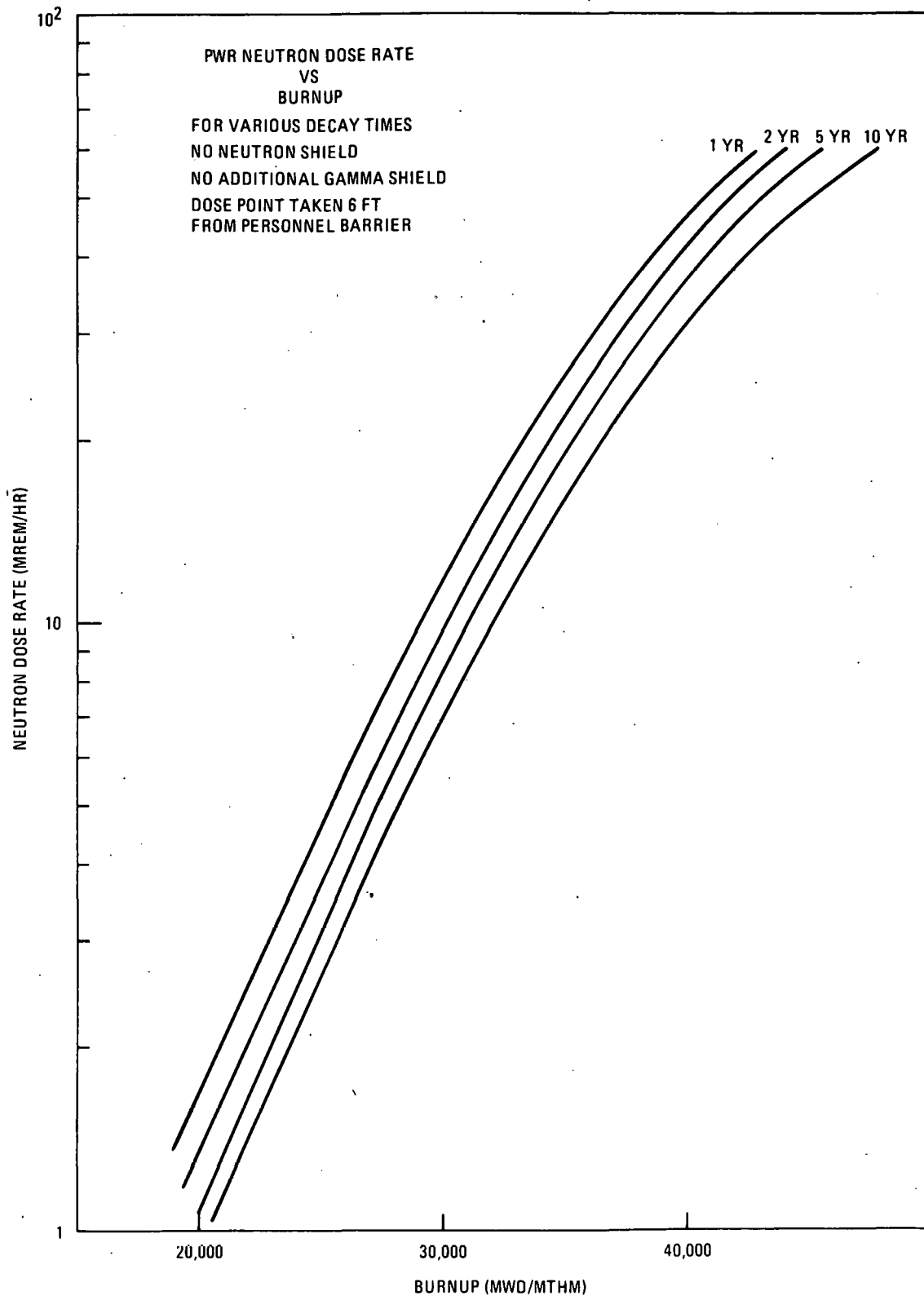
Fig. 3-28. PWR neutron dose rate vs burnup for various decay
times, no neutron shield, no additional gamma
shield
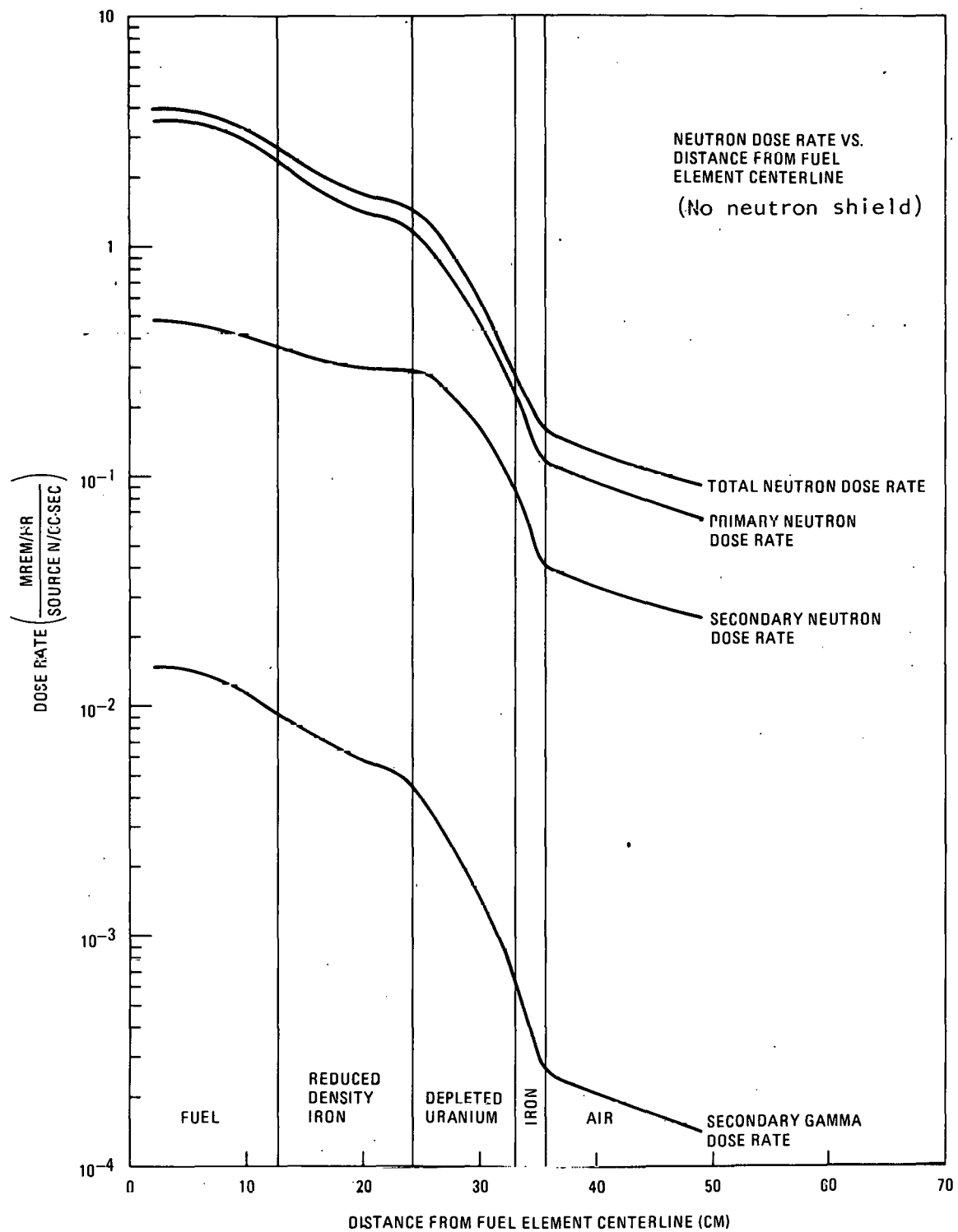
Fig. 3-29. Neutron dose rate vs distance from fuel
element centerline (no neutron shield)

NEUTRON DOSE RATE VS.
DISTANCE FROM FUEL
ELEMENT CENTERLINE
(3 in. neutron
shield)

TOTAL NEUTRON
DOSE RATE

PRIMARY NEUTRON
DOSE RATE

SECONDARY NEUTRON
DOSE RATE

DOSE RATE $\left(\dfrac{\text{MREM/HR}}{\text{SOURCE N/CC-SEC}} \right)$

FUEL

REDUCED
DENSITY
IRON

DEPLETED
URANIUM

IRON

B-Si

AIR

DISTANCE FROM FUEL ELEMENT CENTERLINE (CM)
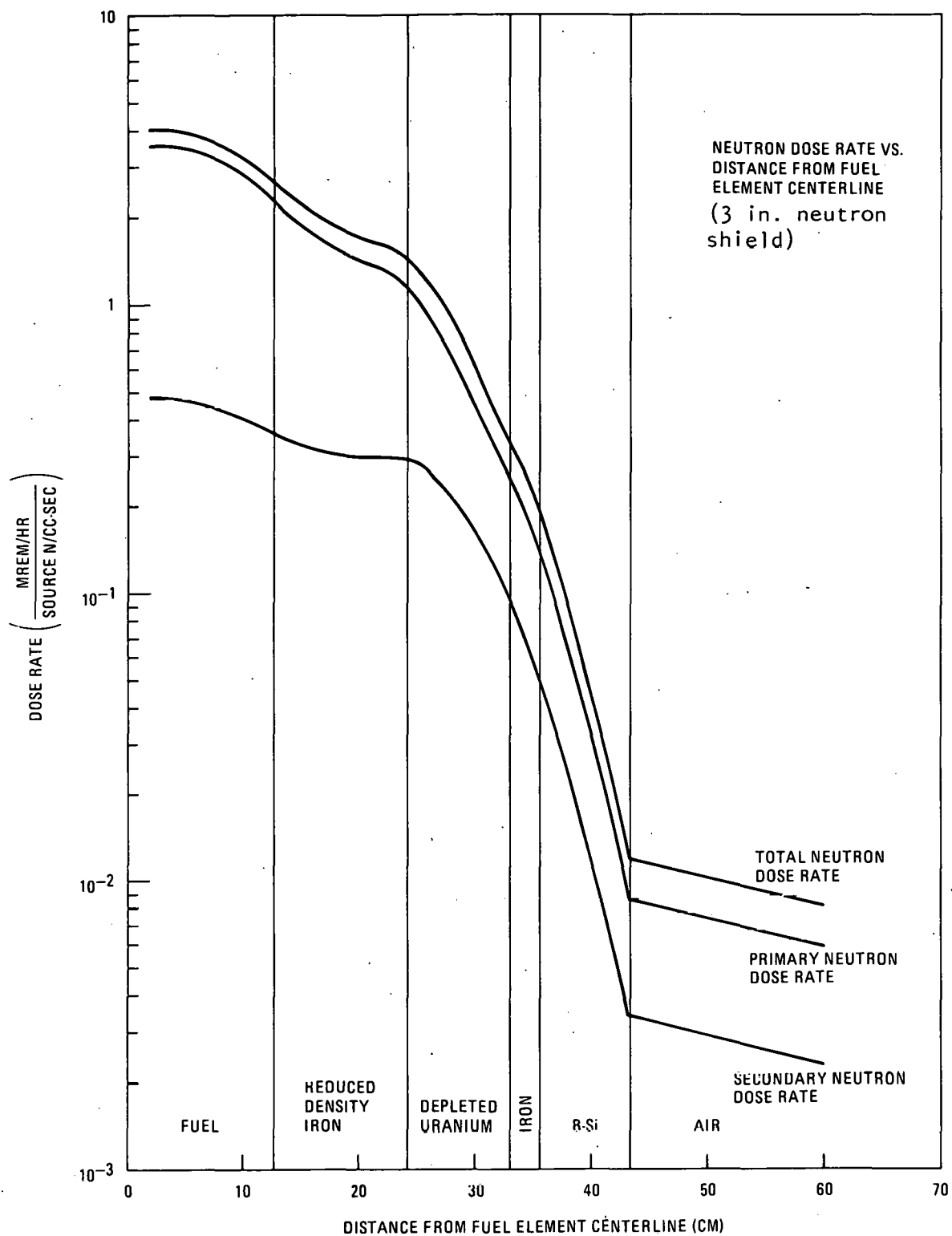
Fig. 3-30.   Neutron dose rate vs distance from fuel
element centerline (3 in. neutron shield)

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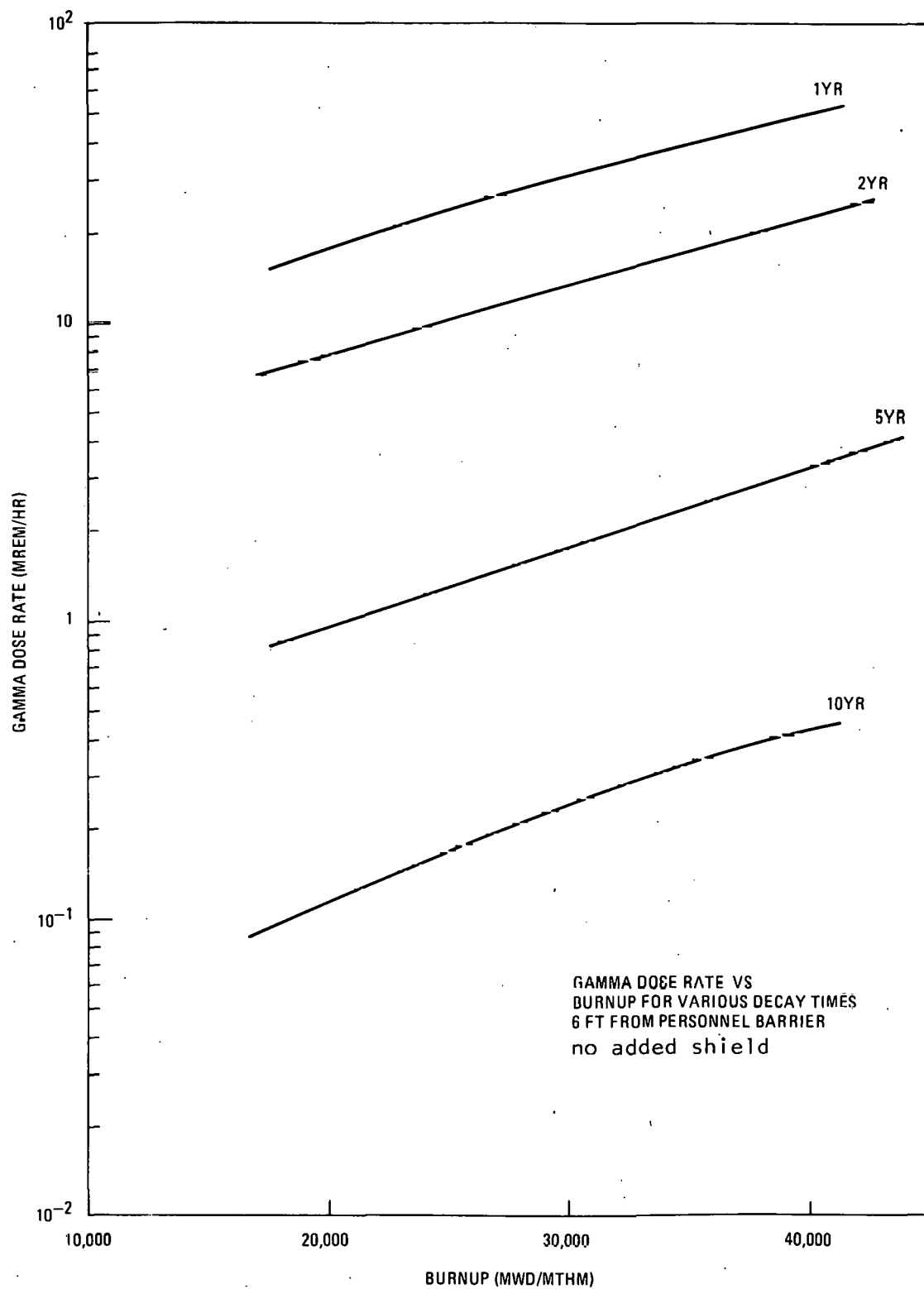
Fig. 3-31. Gamma dose rate vs. burnup for various decay times, no added shield.

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GAMMA DOSE RATE VS
DECAY TIME

FOR VARIOUS BURNUP

6 FT FROM PERSONNEL
BARRIER no added shield
—— PWR
— — BWR

40,000 MWD/MTHM

34,000 MWD/MTHM

33,000 MWD/MTHM

20,000 MWD/MTHM
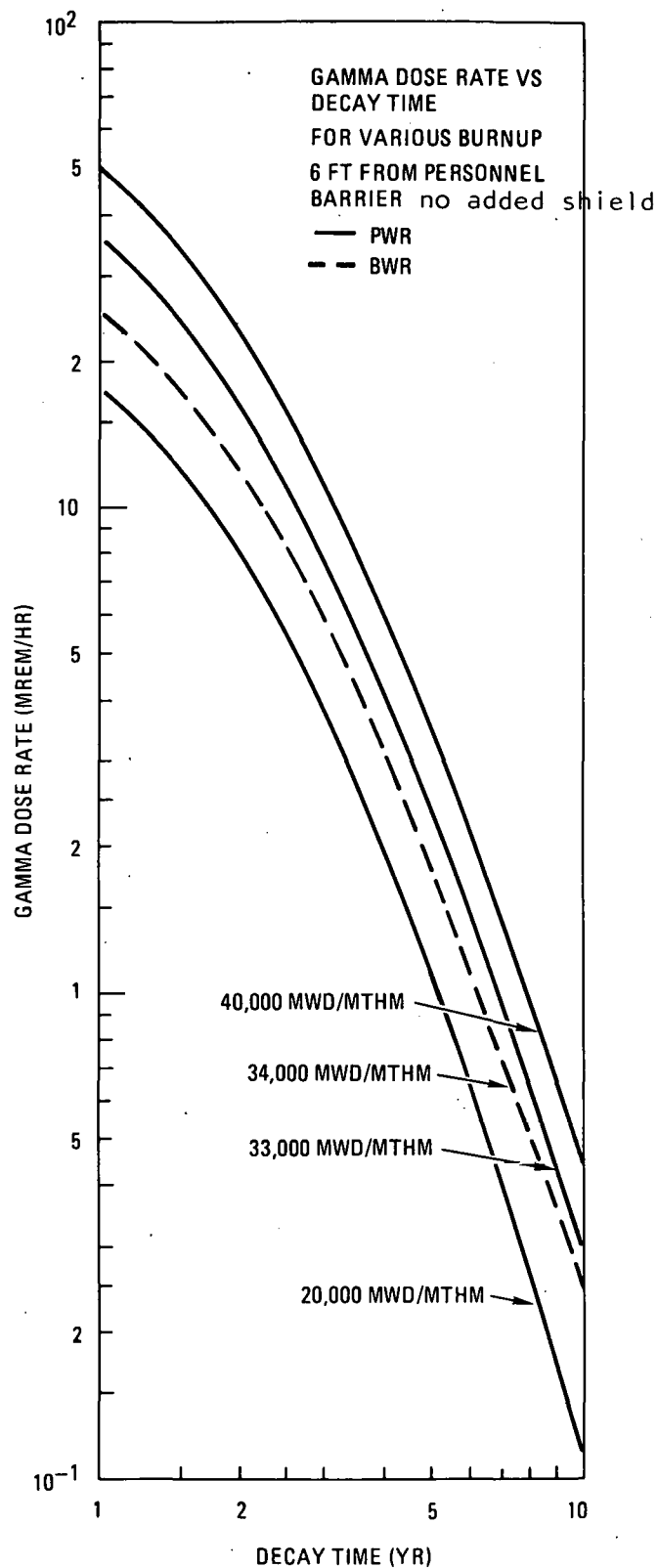
DECAY TIME (YR)

Fig. 3-32.   Gamma dose rate vs decay time for various
burnup,6 ft from personnel barrier , no
added shield.   79

TOTAL DOSE RATE
NORMALIZED TO SIDE OF CASK
VS
DISTANCE AWAY FROM
SIDE OF CASK

PWR FUEL
20,000 MWD/MTHM
1YR DECAY
3 IN. NEUTRON SHIELDING

NORMALIZED TOTAL DOSE RATE

DISTANCE AWAY FROM SIDE OF CASK (M)
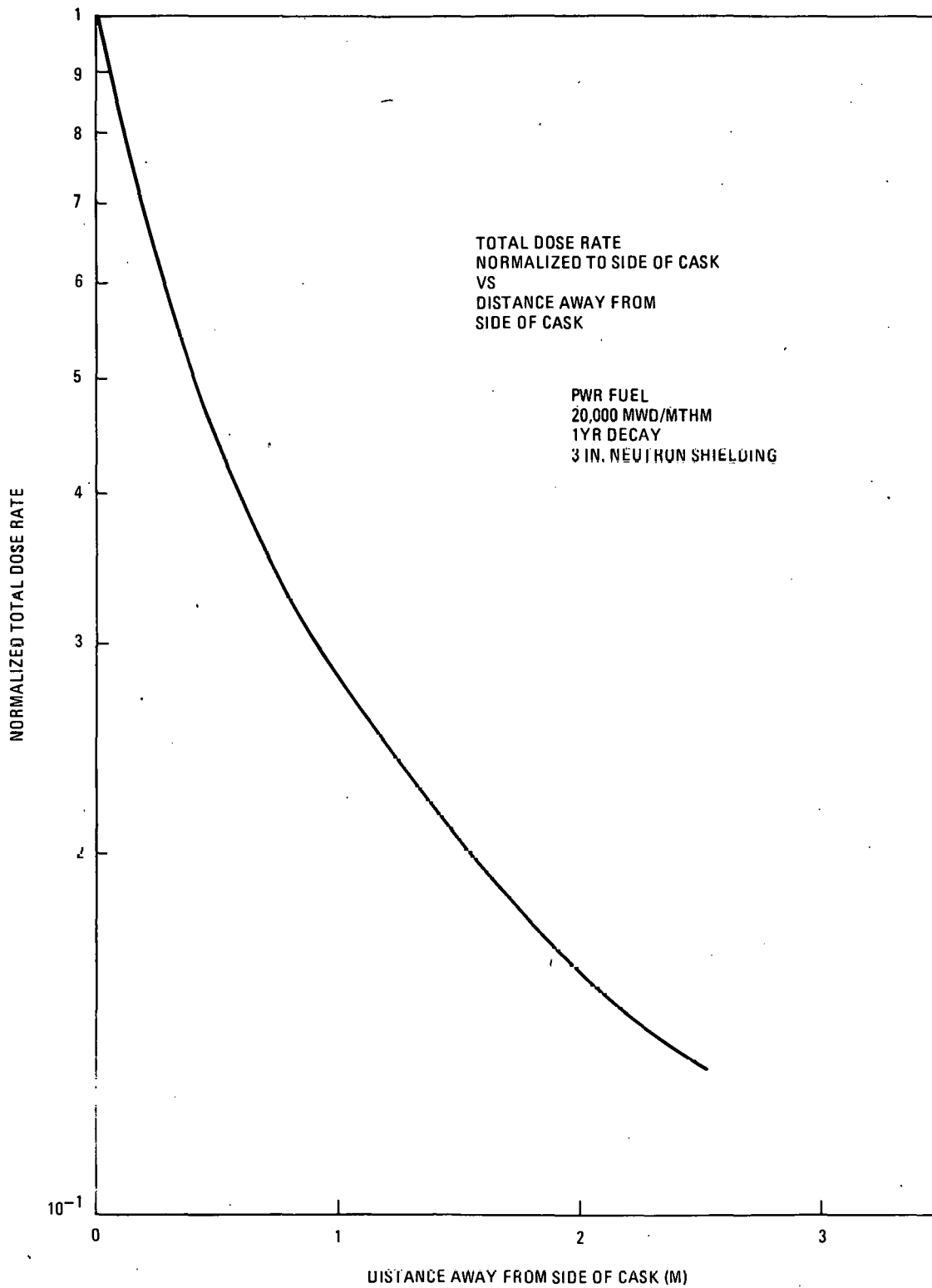
Fig. 3-33.   Total dose rate normalized to side of cask vs
distance away from side of cask

3.5.1.1.3 Total Dose Rates. Total dose rates (neutron + gamma) 6 ft from
the edge of the transport vehicle are plotted against burnup for various
decay times and shielding thicknesses in Figs. 3-34 through 3-39. From
these figures parametric curves of constant shield combinations can be
determined once a maximum dose rate has been set. These curves are
shown in Figs. 3-40 through 3-43 for maximum dose rates of 10, 5, and 2
mrem/hr, respectively. Table 3-13 shows the same data as the figure for
a 10 mrem/hr dose rate criteria but in tabular form. One can see that
the maximum burnup permissible with the as-built cask (no additional
neutron or gamma shielding) for a decay time of 5 yr is 30,000 MWD/MTHM.
If 3 in. of neutron shielding are added, the minimum decay time is
reduced to 2 yr, and so on. From Table 3-13 and Fig. 3-40, one has all
the information necessary for PWR fuel to determine the maximum or mini-
mum value of burnup, decay time, or shield thickness.

The ratio of neutron to gamma dose rates as a function of burnup
is shown in Figs. 3-44 and 3-45 for various shield thicknesses and decay
times.

3.5.1.2 PWR Top Results

Old Design. This concept involved removing the inner linner and lowering
the inner closure 1.25 inches (see section 3.4.2.2). Dose rates, con-
sidering primary radiations only (no secondary neutrons) were calculated
for the outer surface of the inner closure and the adjoining cask body
surfaces. For the results see Table 3-17. Most cases of burnup and
decay result in dose rates that exceed the 50 mrem/hr criteria, fortunately
this concept is no longer being considered.

New Design. This concept uses new inner and outer closures as shown on
Figure 3-21. Dose rates were calculated for the outer surface of the
inner closure and the adjoining cask body surfaces. For the results see
Table 3-18a. Again, most cases of burnup and decay result in dose rates
that exceed the 50 mrem/hr criteria. The addition of gamma and/or neutron
shielding as part of the canister support basket can reduce these dose
rates to satisfy the 50 mrem/hr criteria.

Total dose rates for the same cases of burnup and decay were calculated
for the outer surface of the outer closure. These results are shown in
Table 3-18b.
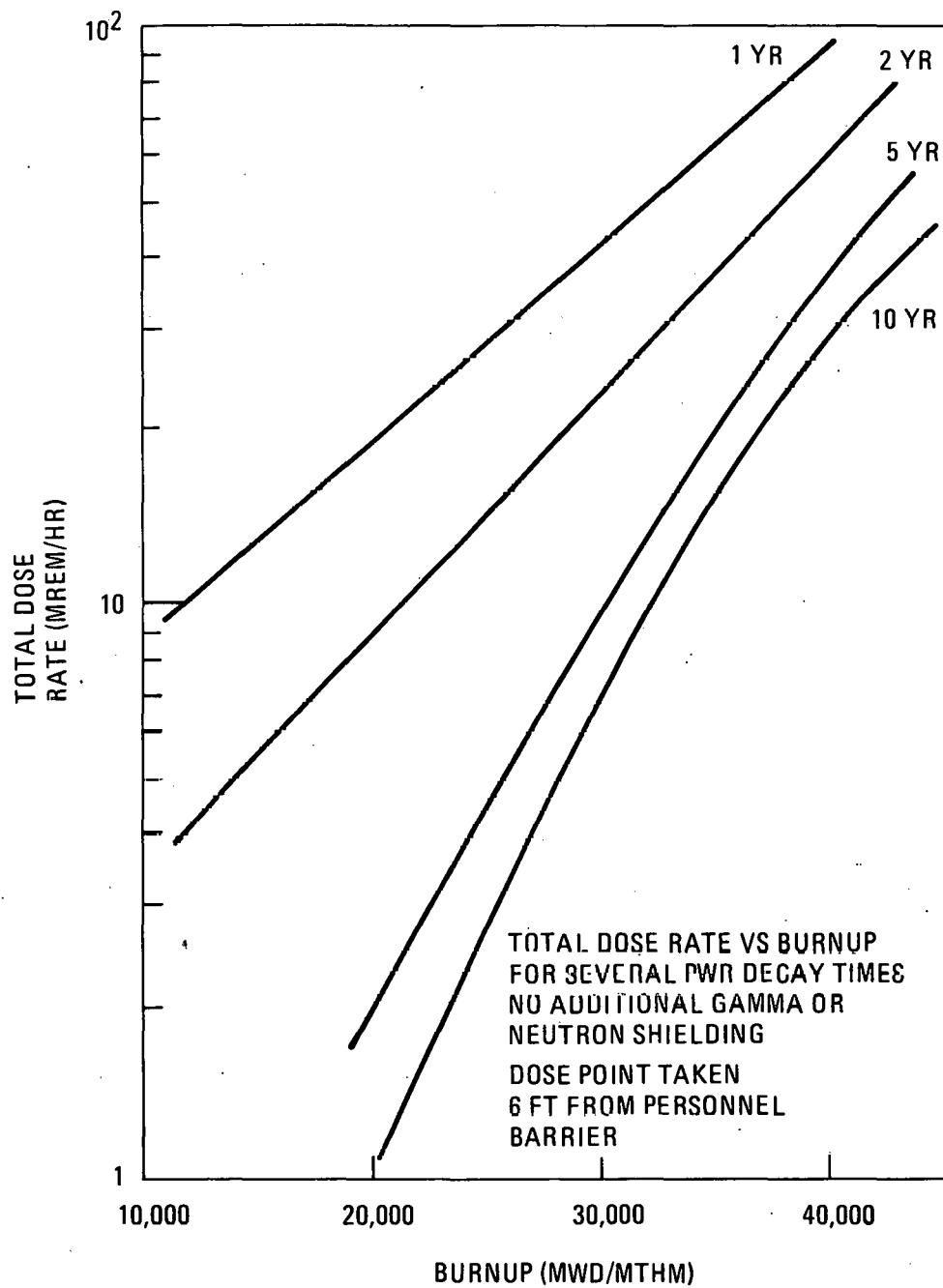
Fig. 3-34. Total dose rate vs burnup for several PWR decay
times, **no** additional gamma shield, **no**
neutron shield
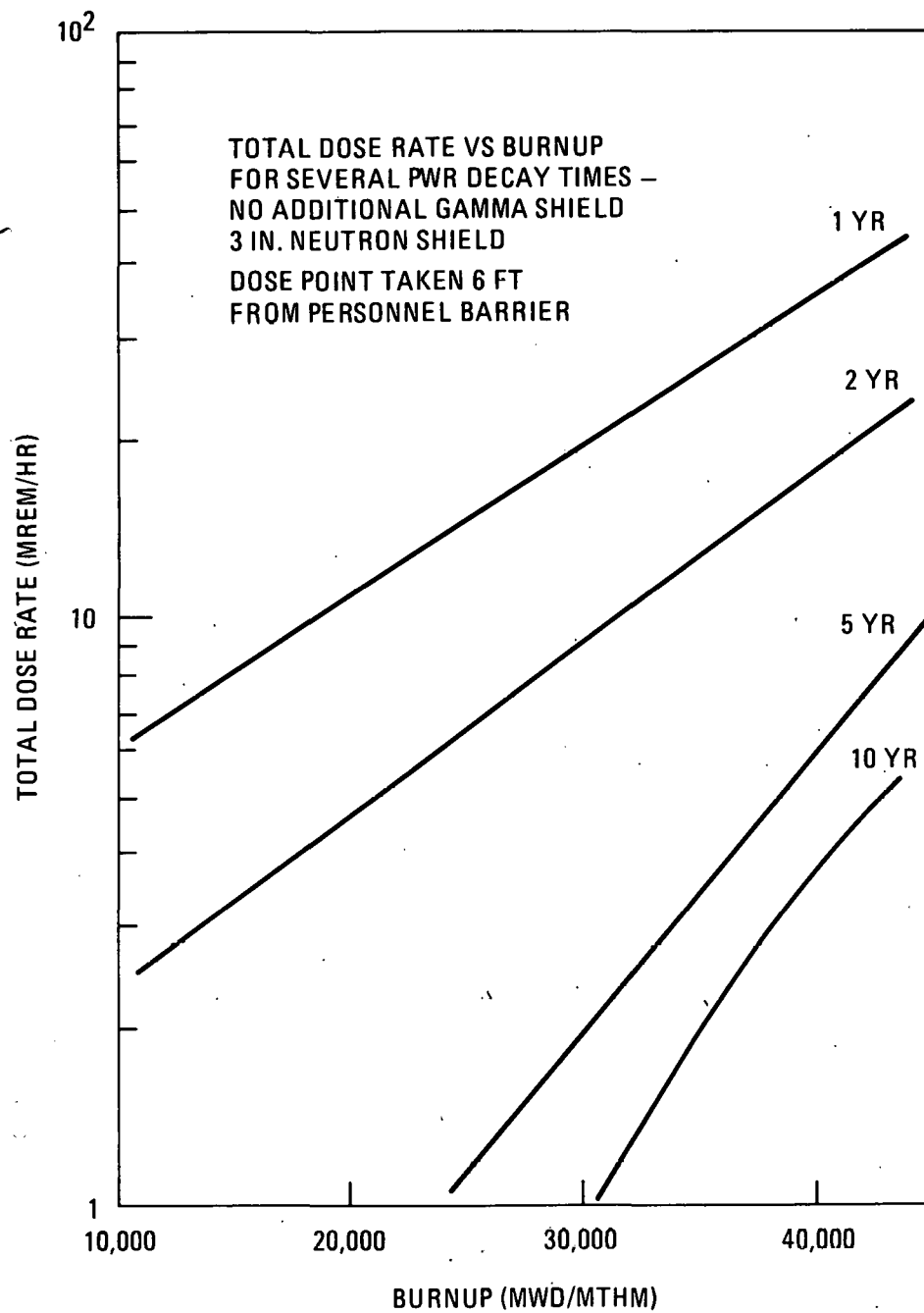
Fig. 3-35. Total dose rate vs burnup for several PWR decay
times, no additional gamma shield, 3 in.
neutron shield

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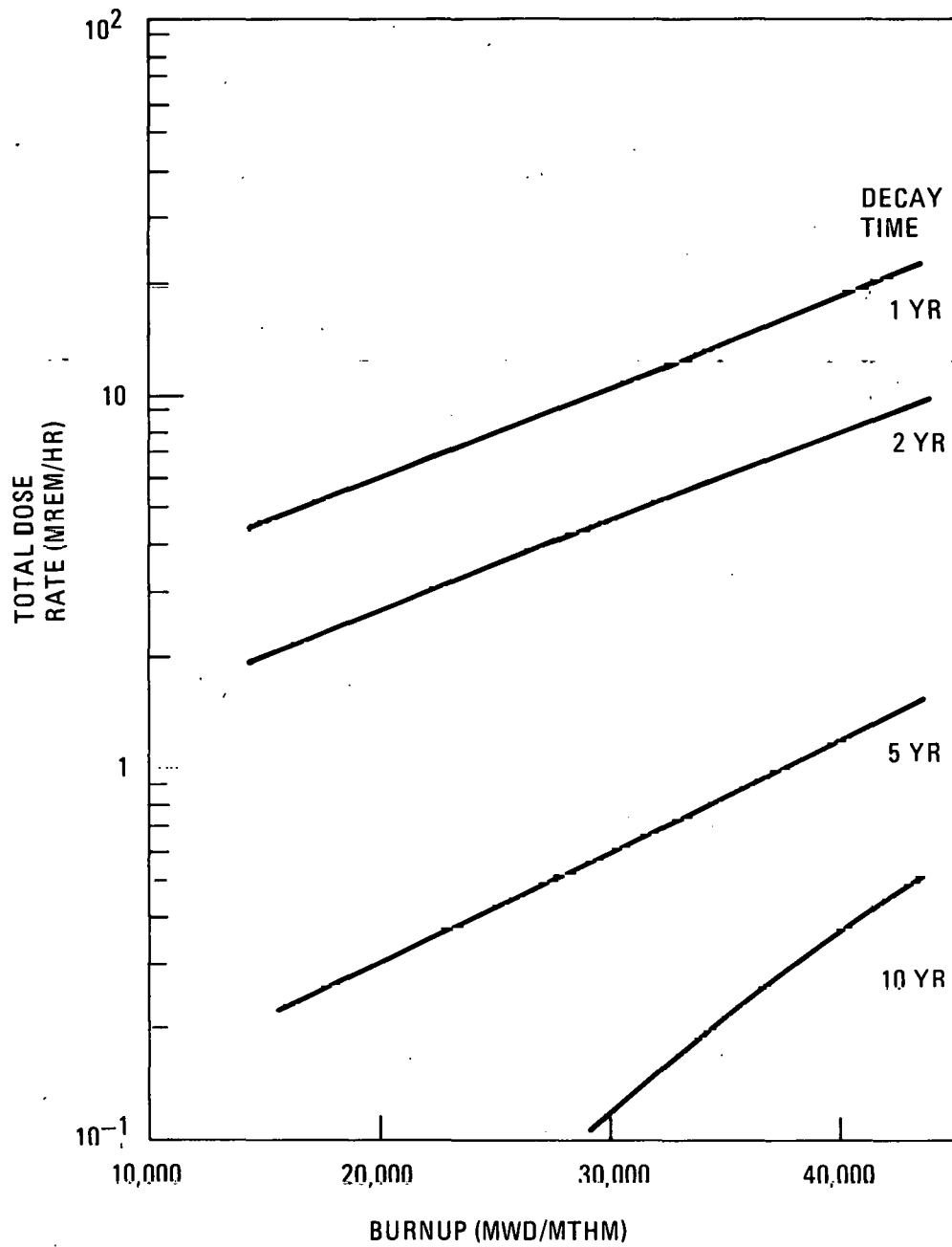
Fig. 3-36. Total dose rate vs burnup for several PWR decay
times, no additional gamma shield, 6 in. neutron
shield
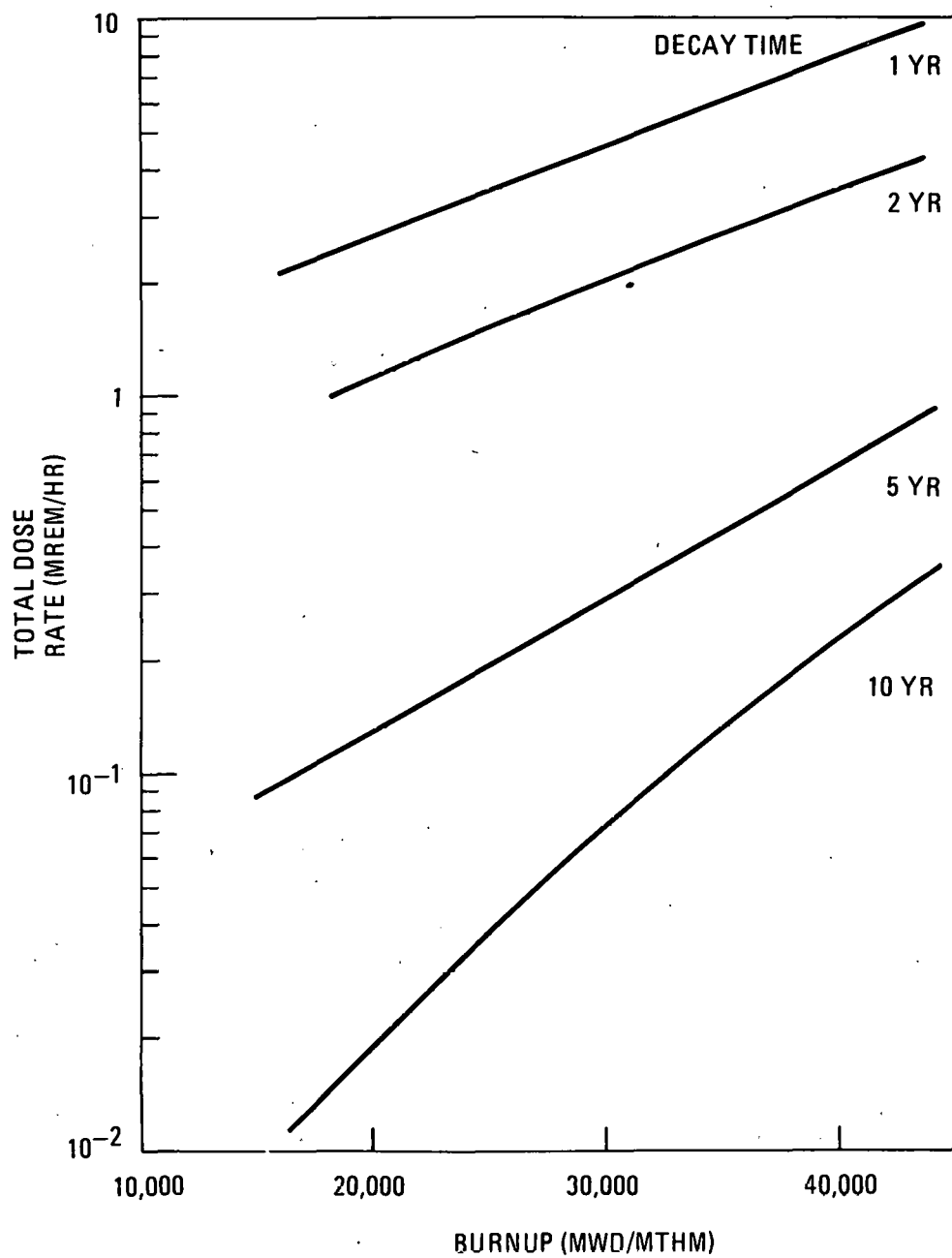
Fig. 3-37. Total dose rate vs burnup for several PWR decay
times 6 ft from personnel barrier, 1 inch additional
gamma shield, 6 inches neutron shield
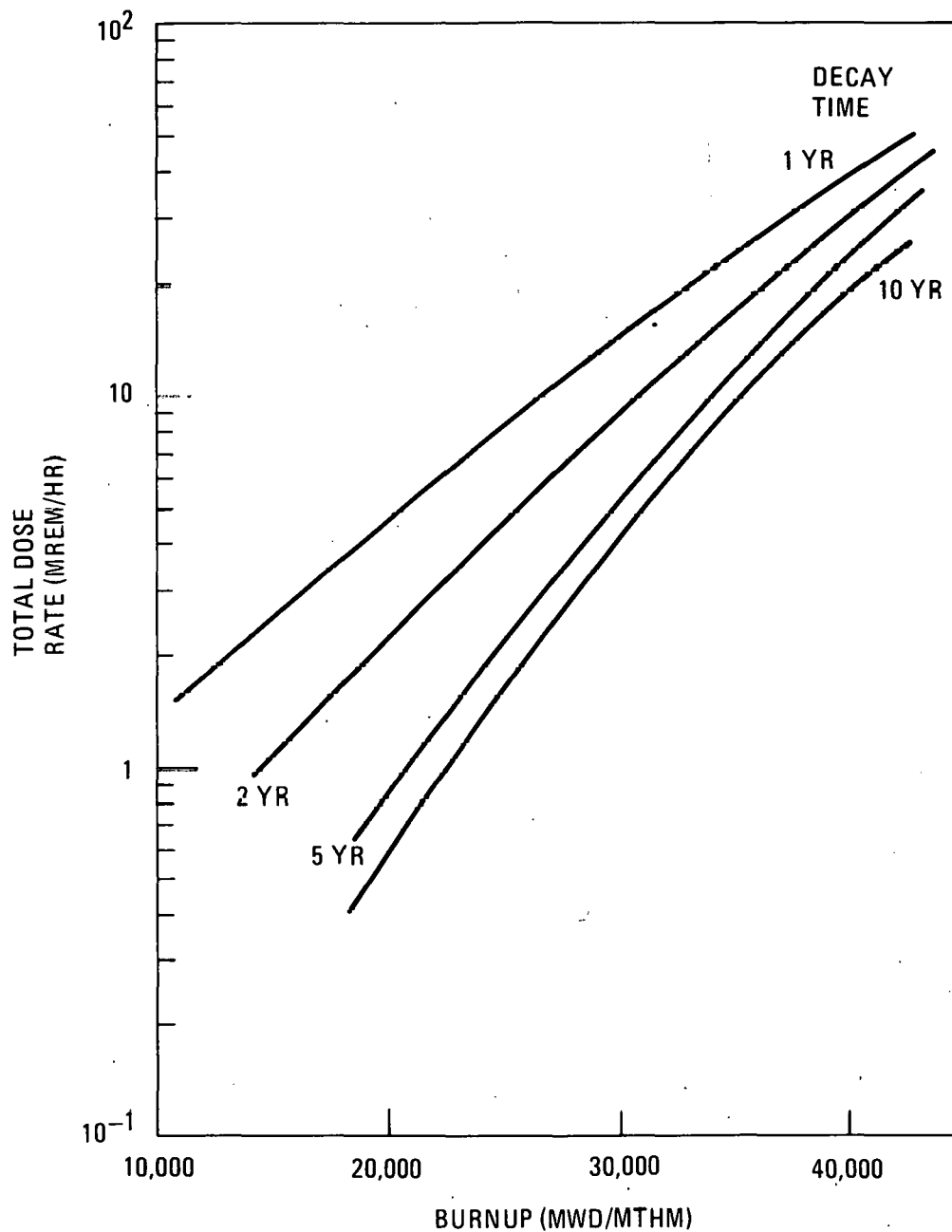
Fig. 3-38. Total dose rate vs burnup for several PWR decay
times, 2 in. additional gamma shield, no neutron
shield
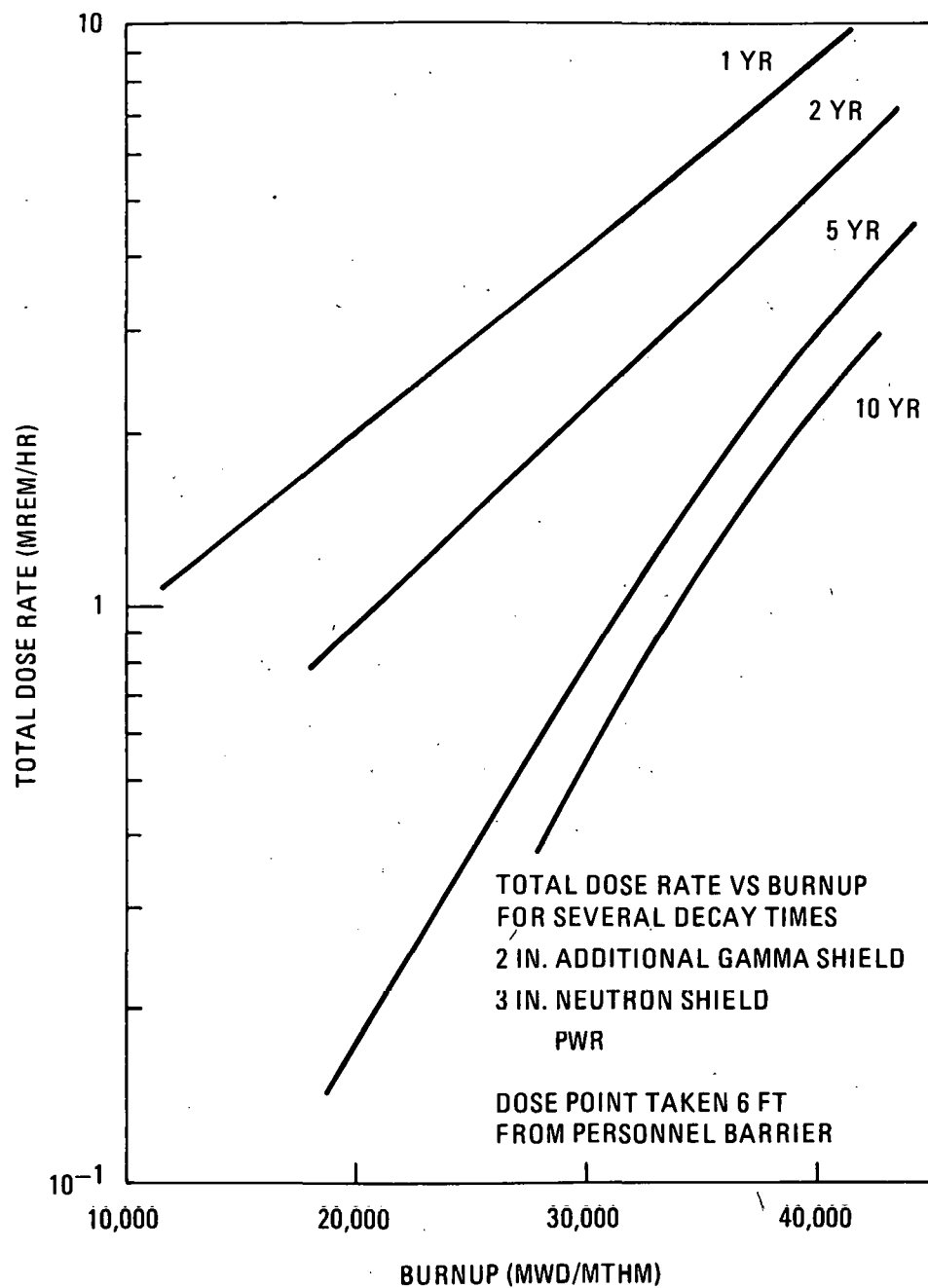
Fig. 3-39. Total dose rate vs burnup for several PWR decay
times – 2 in. additional gamma shield, 3 in.
neutron shield,

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88
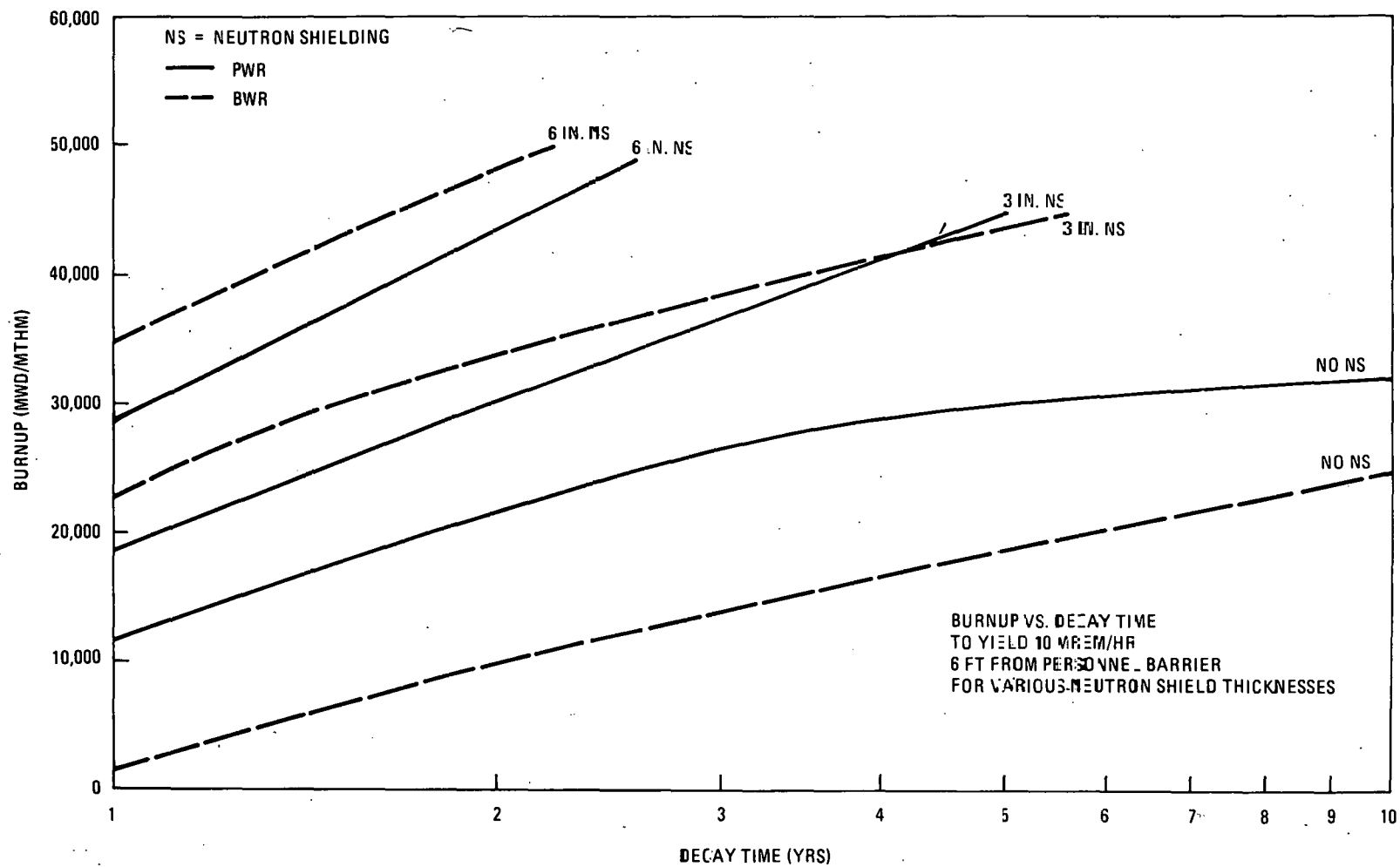
Fig. 3-40. Burnup vs decay time to yield 10 MREM/HR 6 ft from
personnel barrier for various neutron shield
thicknesses
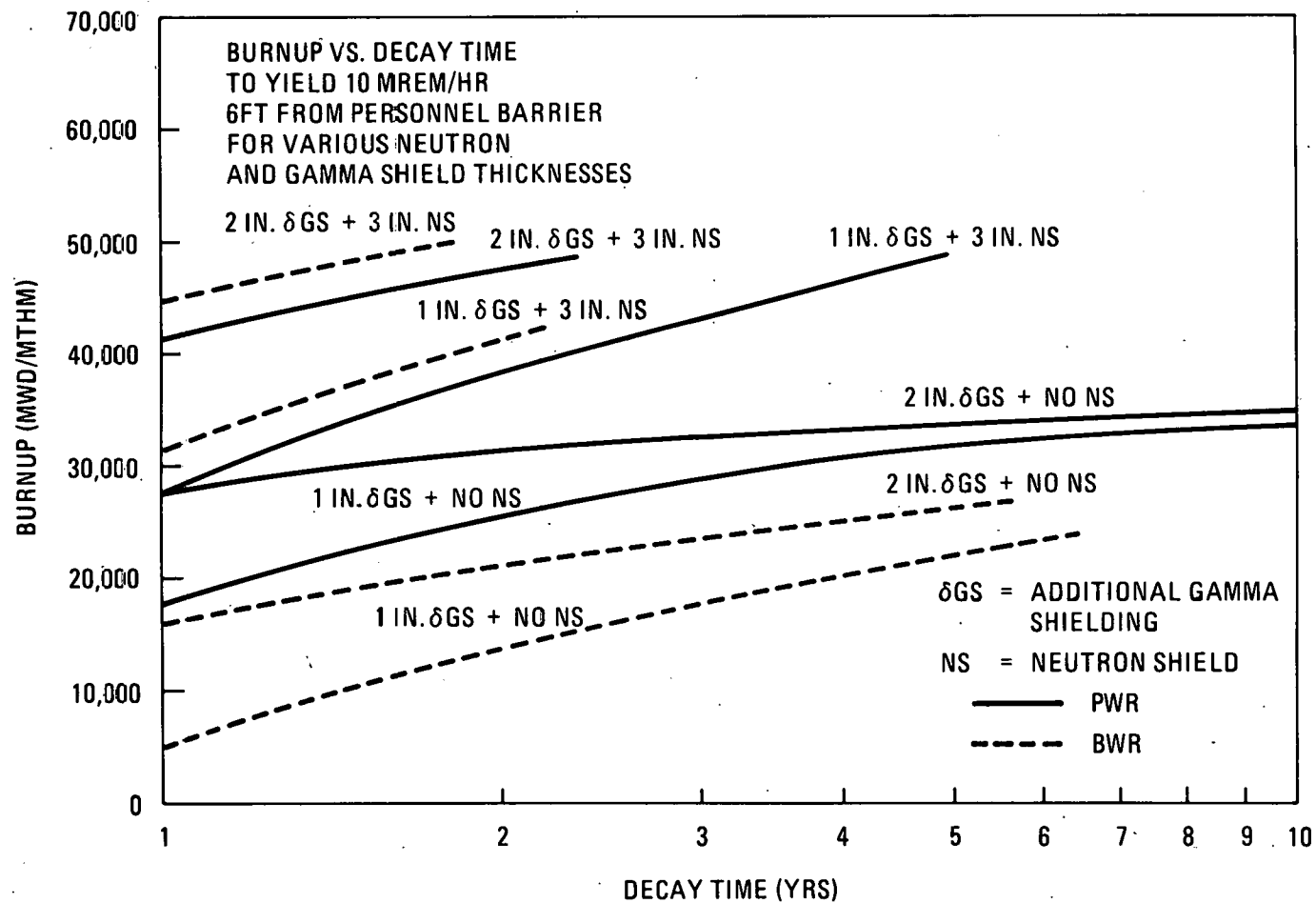
Fig. 3-41. Burnup vs decay time to yield 10 MREM/HR 6 ft from
personnel barrier for various neutron and gamma
shield thicknesses
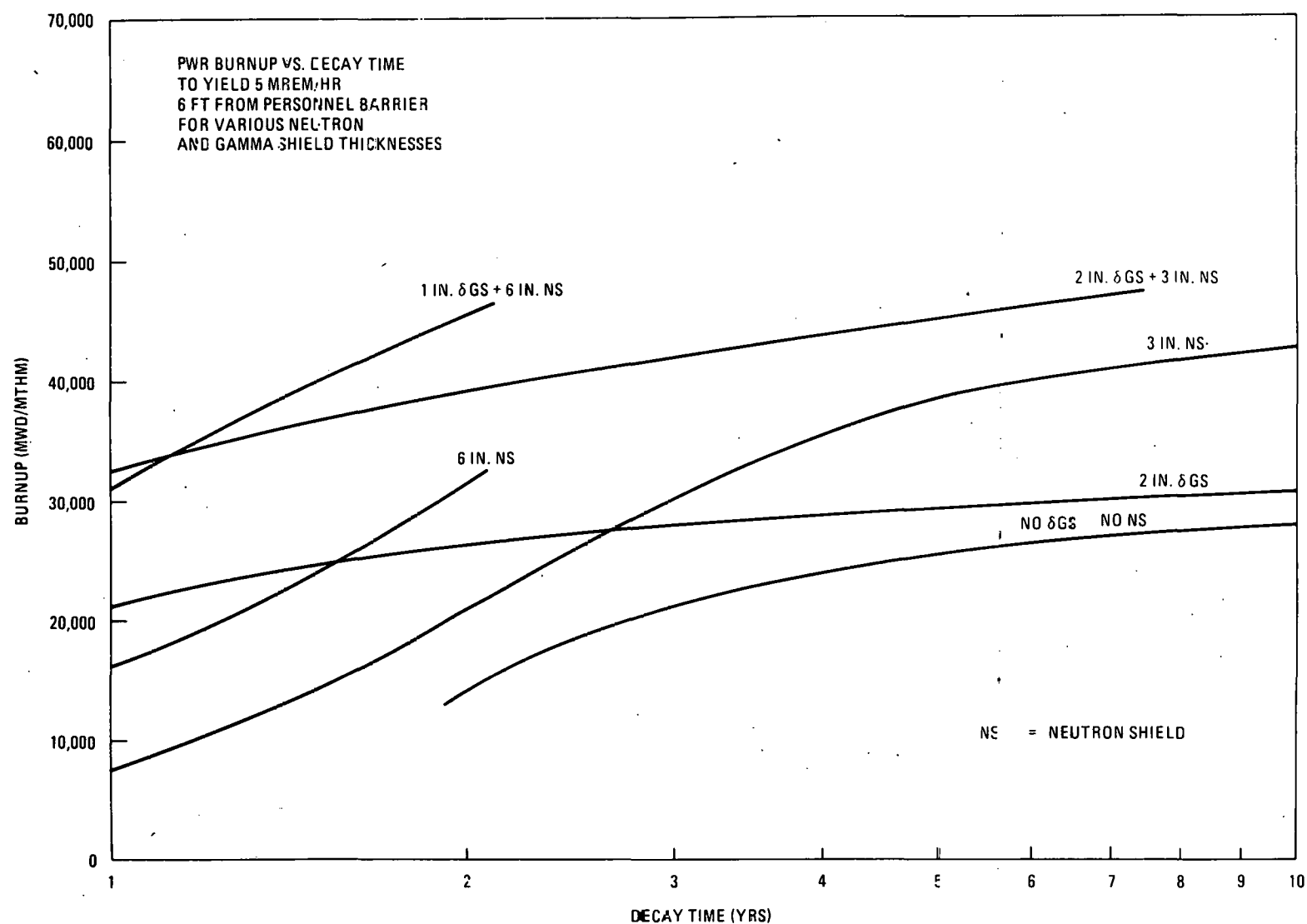
Fig. 3-42. PWR burnup vs decay time to yield 5 MREM/HR 6 ft from
personnel barrier for various neutron and gamma
shield thickness
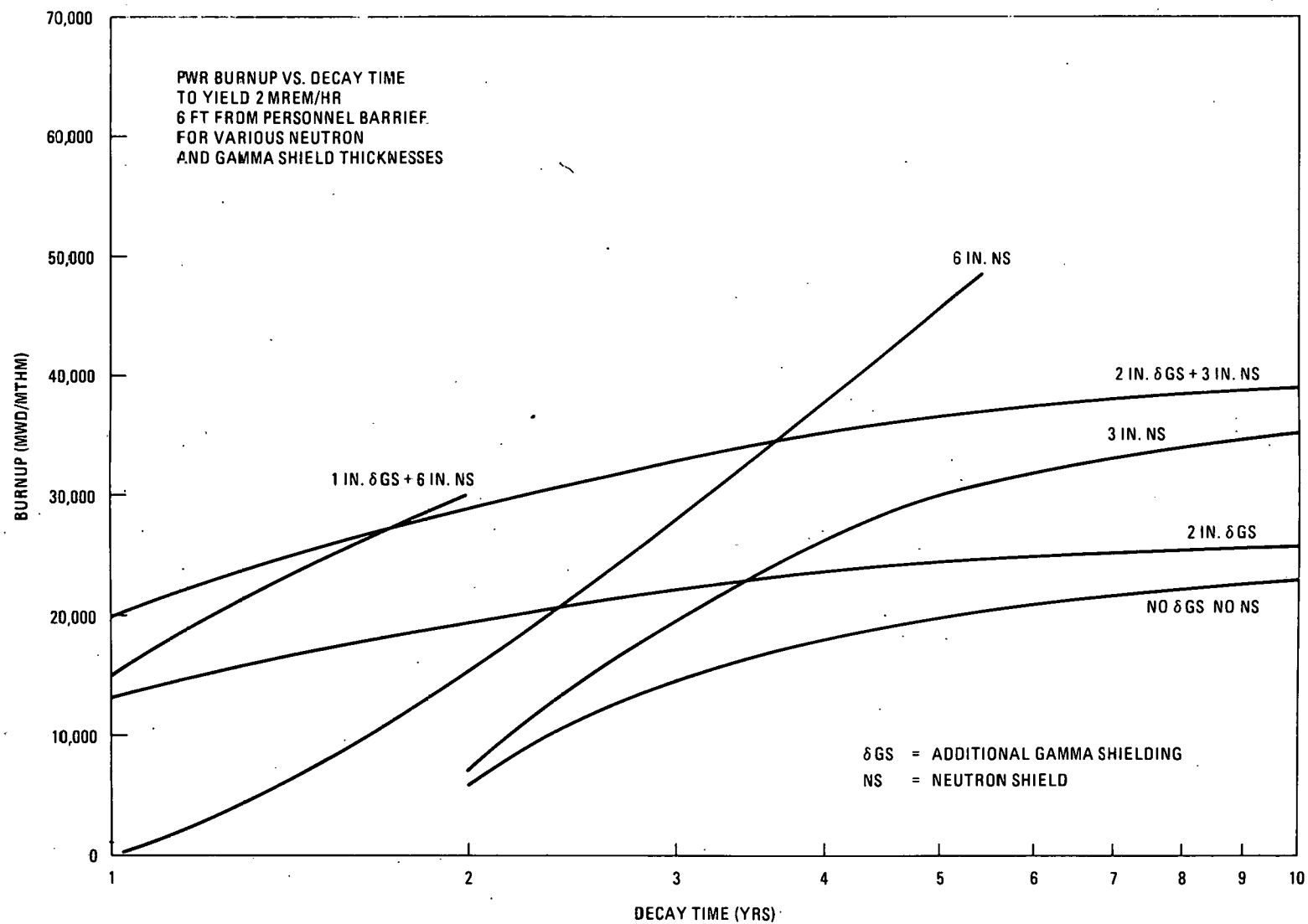
Fig. 3-43. PWR burnup vs decay time to yield 2 MREM/HR 6 ft from personnel barrier for various neutron and gamma shield thicknesses

TABLE 3-13
UNIFORM 10 MREM/HR 6 FT FROM EDGE OF TRANSPORT VEHICLE

Additional Neutron Shield (in.)	Additional Gamma Shield (in.)	Decay Time (yr)	Permissible Burnup (MWD/MTHM)	
			PWR	BWR
0	0	1	11	
		2	21	
		5	30	19
		10	32	25
3	0	1	18.5	18.5
		2	30	34
		5	44.5	43
6	0	1	29	34.5
		2	43.5	48
		5	>50	>50
0	1	1	18	5
		2	25.5	14
		5	32	22.5
		10	33.5	28
3	1	1	27.5	31.5
		2	38.5	41.5
		5	49	>50
6	1	1	44	>50
		2	>50	>50
0	2	1	27.5	16
		2	31	21
		5	34	26.5
		10	35	31
3	2	1	41	44.5
		2	47.5	>50
		5	>50	>50
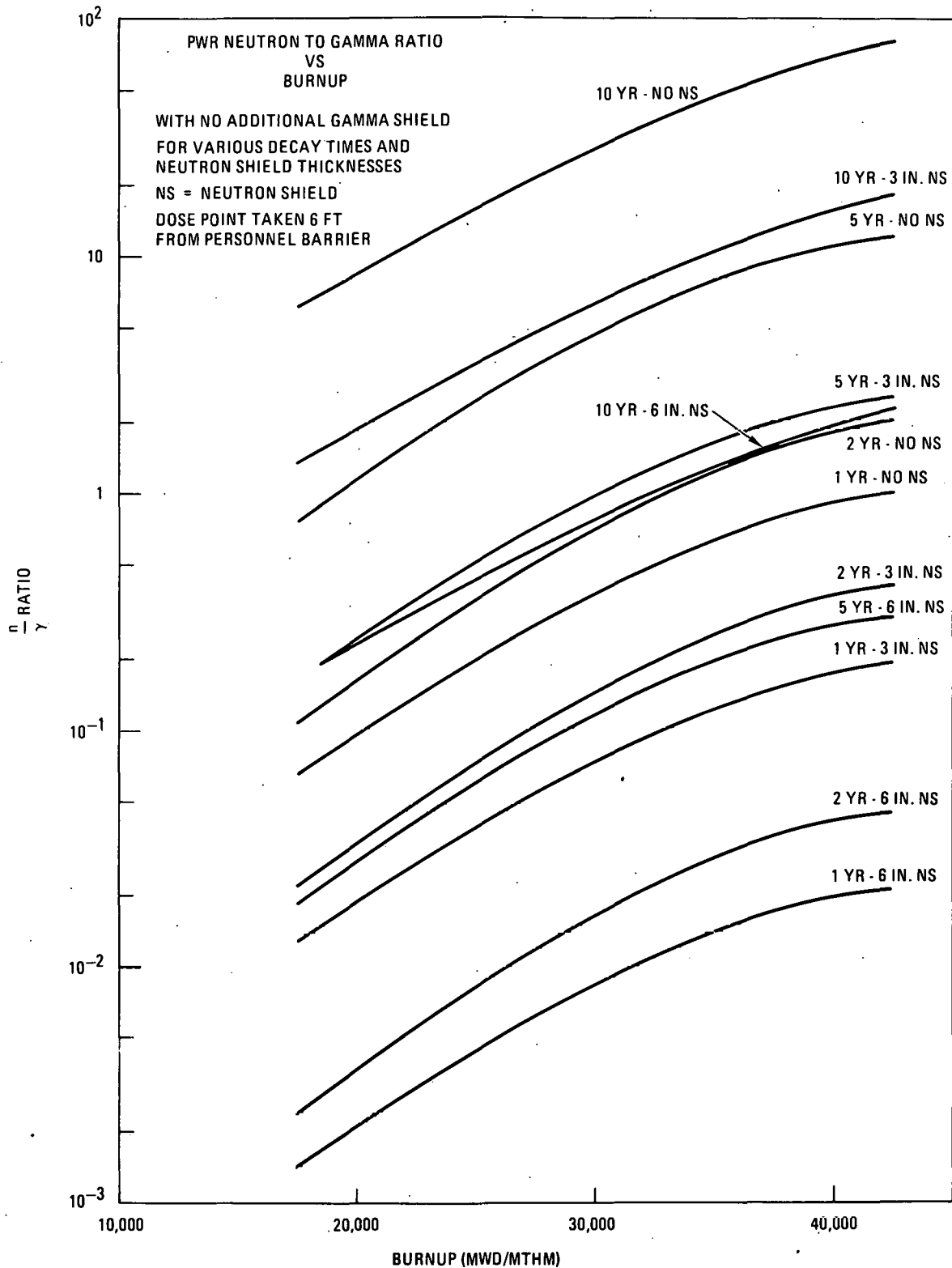
Fig. 3-44.   PWR neutron to gamma ratio vs burnup with no
             additional gamma shield for various decay
             times and neutron shield thicknesses

93
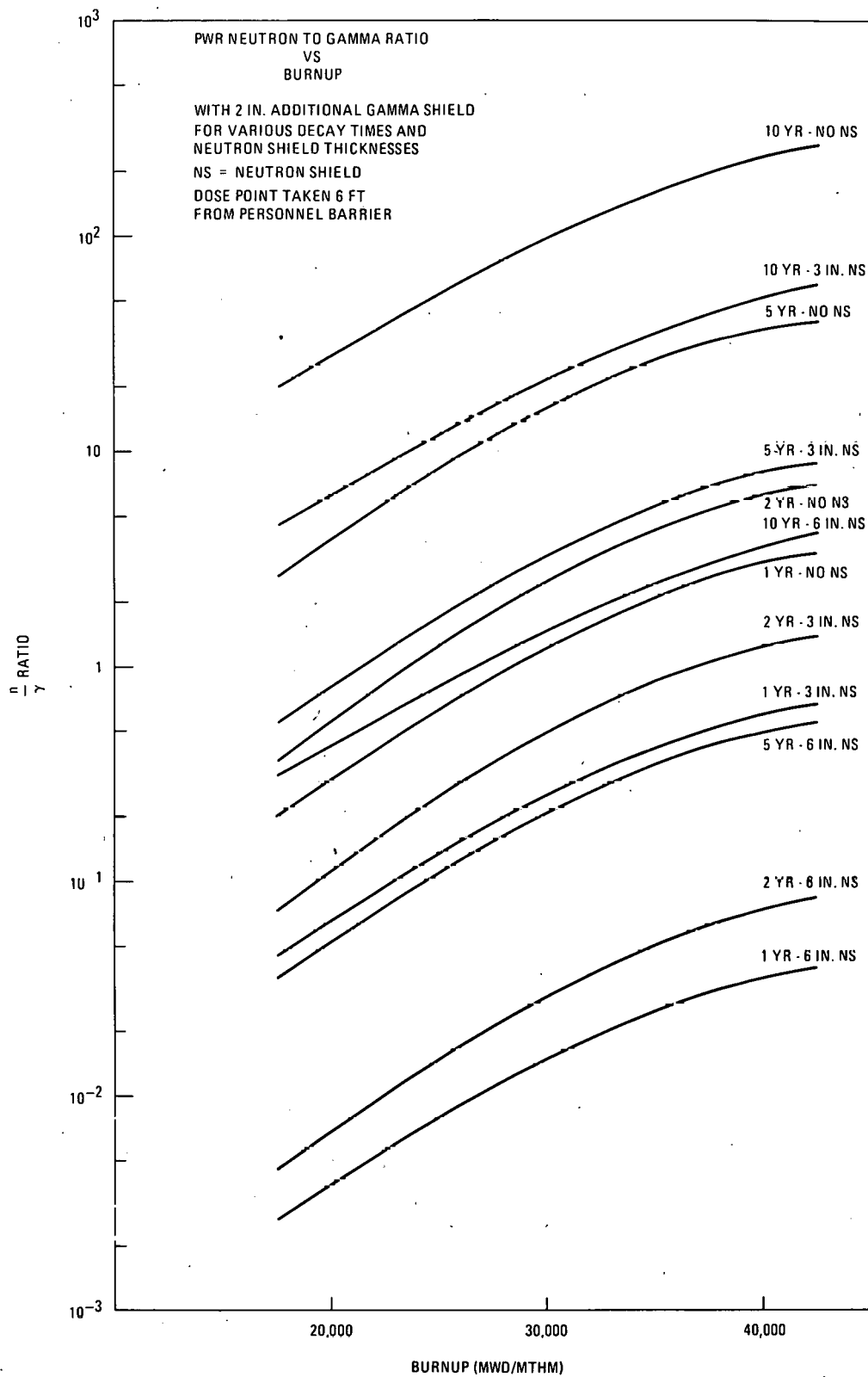
Fig. 3-45. PWR neutron to gamma ratio vs burnup with 2 in.
additional gamma shield for various decay times
and neutron shield thicknesses

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TABLE 3-17
TOTAL DOSE RATE AT SURFACE OF INNER
CLOSURE - OLD DESIGN
mrem/hr

Burnup (MWD/MTHM)	Decay Time			
	1 yr	2 yr	5 yr	10 yr
20,000	230	119	33.6	14.5
33,000	528	308	128	74
40,000	819	510	252	161

TABLE 3-18a
TOTAL DOSE RATE AT SURFACE OF INNER
CLOSURE - NEW DESIGN
mrem/hr

Burnup (MWD/MTHM)	Decay Time			
	1 yr	2 yr	5 yr	10 yr
20,000	54.9	27.6	8.1	4.4
33,000	163	102	56.5	41.5
40,000	296	206	136	104

TABLE 3-18b
TOTAL DOSE RATE AT SURFACE OF OUTER
CLOSURE - NEW DESIGN
mrem/hr

Burnup (MWD/MTHM)	Decay Time			
	1 yr	2 yr	5 yr	10 yr
20,000	1.0	0.8	0.7	0.6
33,000	11.8	10	8.7	7.3
40,000	29	26	23	19

3.5.1.3 PWR Bottom Results

Dose rate results from the bottom were calculated assuming that the bottom of the inner container remained within the cask. These results are shown in Table 3-14 for a dose point location 6 ft from the rear of the transport vehicle (154 in. from the bottom of the cask).

An extra 3 inches of gamma shield will be required for the bottom to insure that the surface dose rate does not exceed 200 mrem/hr for the worst case of 40,000 MWD/MTHM burnup fuel with 1 yr. decay. With this additional 3 inches of gamma shielding on the bottom, the new dose rates located 6 ft from the rear of the transport vehicle are shown in Table 3-16. Dose rates without this additional shielding are shown on Table 3-15.

3.5.2 BWR

Neutron and gamma dose rate results for the BWR case are limited, since only one value of burnup was considered, (34,000 MWD/MTHM). Because of the large extrapolations necessary, these results are to be taken with only general trends in mind. No conclusions should be drawn because the errors involved may be large.

3.5.2.1 Side

3.5.2.1.1 Neutrons

Neutron dose rates were calculated to a distance of 6 ft from the edge of the transport vehicle. The results are plotted in Fig. 3-27 against the fuel assembly decay time for the case of no additional gamma shield. Neutron dose rates for other gamma shield thicknesses were calculated using the method described in Section 3.2.2.

3.5.2.1.2 Gammas

Gamma dose rates 6 ft from the edge of the transport vehicle are plotted against decay time in Fig. 3-32 for a burnup of 34,000 MWD/MTHM.

3.5.2.1. 3 Total Dose Rates.

The method for determining parametric relationships between all of the variables considered in this report does not hold when there is only one burnup considered, as in the BWR case. Therefore, an assumption has been made and applied which may have large errors involved but nonetheless gives an insight into the allowable burnups and decay times required. It has been assumed that when plotting the total dose rate versus burnup for constant decay time the resulting

TABLE 3-14
TOTAL DOSE RATE – m rem/hr – 6 FT FROM END OF TRUCK[a]

Burnup (MWD/MTHM)	1 yr	2 yr	5 yr	10 yr
20,000	1.11	0.5	0.18	0.11
33,000	3.86	2.55	1.59	1.24
40,000	7.51	5.54	3.98	3.17

[a] Includes factors 1.1 for gamma and 1.4 for neutrons.

TABLE 3-15

TOTAL DOSE RATE AT BOTTOM OF CASK[a] MREM/HR

Burnup (MWD/MTHM)	1 yr	2 yr	5 yr	10 yr
20,000	93.8	42.2	15.2	9.3
33,000	326.2	215.5	134.4	104.8
40,000	634.6	468.1	336.3	267.9

[a] 84.5 X dose rate 6 Ft. from truck (Table 3-14)

TABLE 3-16

DOSE RATES 6 FT FROM REAR OF TRUCK DUE TO 3 IN. OF ADDED
GAMMA SHIELD ON BOTTOM*

m rem/hr

(MWD/MTHM) Burnup	1 yr	2 yr	5 yr	10 yr
20,000	0.13	0.08	0.06	0.05
33,000	1.03	0.84	0.70	0.58
40,000	2.43	2.12	1.82	1.51

*154 inches from bottom of cask

curve for BWR data has the same slope as for PWR data. With this in mind, the BWR parametric results are presented in Table 3-13 and figs. 3-40 and 3-41 for the case of 10 mrem/hr at 6 ft from the edge of the transport vehicle.

3.5.2.2 BWR Top and Bottom

The dose rates due to BWR furl with a burnup of 34,000 MWD/MTHM will be about 70% of those due to PWR fuel with 33,000 MWD/MTHM burnup.

3.5.3 PWR/BWR Comparison

For the same burnup, the BWR fuel assembly is characterized by a volumetric neutron source 60% higher than that of the PWR fuel, primarily as a result of the lower U-235 enrichment in BWR fuel and the consequent higher neutron absorption rate in U-238 (eventually leading to curium). Although the BWR gamma source is 30% lower than the PWR source, the neutron difference has the most impact on the relative suitability of the FSV-1 cask to carry LWR fuel, particularly if no neutron shielding is added. Thus, the summary Table 3-13 shows that with no neutron shielding and a decay time of 2 yr., the cask can carry PWR fuel with a burnup of 21,000 MWD/MTHM. Three more years decay time is needed to carry the BWR fuel with the same burnup. With a thick neutron shield, the positions of the PWR and BWR reverse with the cask being able to carry slightly higher BWR burnups.

One other obvious aspect should be pointed out in a PWR/BWR comparison. If the BWR dose rates are too high, one (instead of three) assemblies could be transported, whereas there is no such "out" for PWR fuel.

Finally, it should be emphasized that the results for BWR fuel reported herein are based on only one ORIGEN case, namely for 34,000 MWD/MTHM burnup, and that some of the large extrapolations from this one case lead to larger uncertainties in BWR results than in PWR results. The limiting BWR burnups for no additional shielding are particularly crude numbers.

3.5.4 Comparison with Earlier Results

The earlier work mentioned in the Introduction (Ref. 3-8) can be compared against the present results. Gamma dose rates are compared in Fig. 3-46. The rather wide divergence is apparently due to a difference in source terms between Ref. 3-8 and the SANDIA-ORIGEN output, as shown in Table 3-19.

GA experience has been that photon emission at energies greater than 2.2 MeV is completely negligible. This conclusion is borne out by a recent study performed by SAI for EPRI (Ref. 3-19). At 1 yr decay the LWR fission product spectrum is given as 3.9×10^{-4} MeV/fission-group for 1.8-2.2 MeV photons and only 4.4×10^{-13} MeV/fission-group for the >2.6 MeV group.

Since for the worst cases the ORIGEN-generated 2.75-MeV source group contributes 60% of the total gamma dose rate, a large positive error in this group greatly penalizes the predicted performance of the cask. Future studies should be directed at improving the accuracy of the SANDIA-ORIGEN gamma spectrum and achieving consistency with that published by other authors.

Table 3-20 provides a comparison between neutron results from Ref. 3-8 and the present study. In spite of the fact that the SANDIA-ORIGEN calculations utilized an apparently high value of the (α,n) conversion, the present neutron results show higher permissible burnup than the Ref. 3-8 results. The reason for this reverse comparison can be sought in the original source terms used in each study, as shown in Table 3-21.

However, this table also shows the differences in the wrong direction. It is tentatively concluded, therefore, that the neutron attenuation and dose estimates in Ref. 3-8 were on the conservative side.
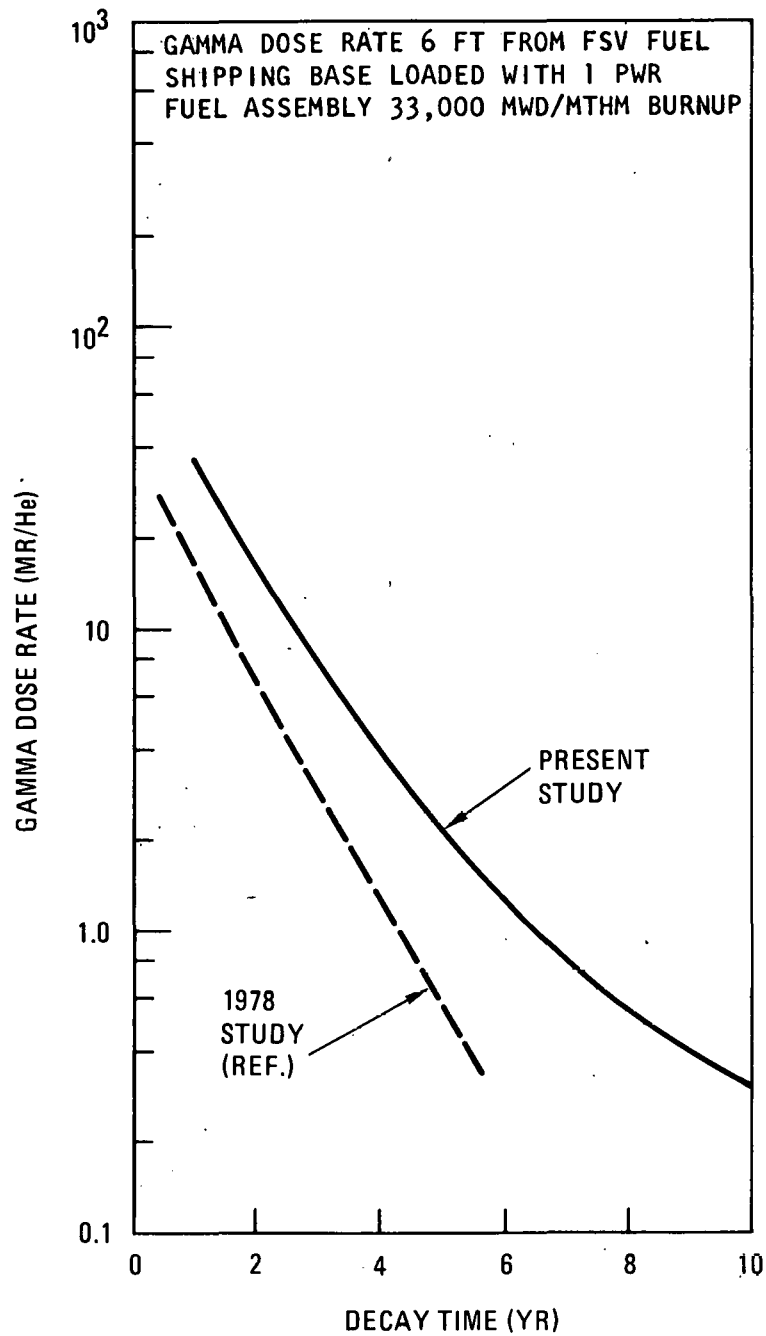
Fig. 3-46. Gamma dose rate 6 ft from FSV fuel shipping
cask loaded with 1 PWR fuel assembly 33,000
MWD/MTHM burnup

TABLE 3-19
PWR GAMMA SOURCE STRENGTH COMPARISON
(MeV/cc-sec)

Gamma Energy (MeV)	1 yr Decay		5 yr Decay		10 yr Decay	
	Ref. 3-8	SANDIA	Ref. 3-8	SANDIA	Ref. 3-8	SANDIA
1.8-2.2	1.02+9	7.0+8	3.21+7	2.1+7	4.71+5	2.96+5
2.5-3.0	Nil	3.6+8	Nil	6.78+6	Nil	7.58+4

TABLE 3-20
NEUTRON SOURCE TERM COMPARISON PWR FUEL

Burnup (MWD/MTHM)	Decay Time (yr)	neutrons/sec-g U	
		Ref. 3-8	Present Study[a]
20,000	1	65	66
	2	40	50
	10	25	37
33,000	1	600	730
	2	500	625
	10	350	440
40,000	1	1500	1800
	2	1300	1600
	10	900	1170

[a] Includes secondary neutrons.

TABLE 3-21

FSV CASK NEUTRON RESULTS COMPARISON

(BURNUP IN MWD/MTHM TO MEET STATED CONDITIONS)

ONE PWR ELEMENT

	10 mrem/hr at 6 ft, No Neutron Shielding or 2 mrem/hr at 6 ft with 2-in. Neutron Shielding Decay (yr)			10 mrem/hr at 6 ft with 2-in. Neutron Shielding or 2 mrem/hr at 6 ft with 4-in. Neutron Shielding Decay (yr)		
	1	2	10	1	2	10
Ref. 3-8	22,500	24,500	26,500	32,000	33,500	36,000
Present study	29,000	30,000	32,000	43,000	44,000	48,000

3.5.5 Uncertainties

Uncertainties in an analysis such as this can be significant and can be both of the systematic and random types. Eventual dose rate measurements of the FSV-1 cask containing LWR fuel may closely agree with predictions only because of compensating errors. This study would be incomplete without at least a qualitative assessment of the sources and magnitudes of errors.

3.5.5.1 Source Terms - gamma. The assumption that Ag110m emits a 2.75 MeV photon 100% of the time caused an overestimation of gamma dose rates. It is possible that numerous other gamma source term errors exist, either in gamma decay spectra, half lives, branching ratios, fission yields, or the actual burnup calculation. It is estimated that the gamma source terms at 1 year decay are uncertain by $\pm$ 50% (after correcting for Ag 110m), and at 10 years decay are uncertain by $\pm$ 75%.

3.5.5.2 Source Terms - neutron. The (α, n) neutron intensity can be uncertain by a factor of 2, since published data in literature displays that much variation. The spontaneous fission rates are much better known.

The inventories of neutron-producing actinides are also uncertain, especially with the one-group burnup calculation used in the SANDIA-ORIGEN code.

The net uncertainty in neutron source is believed to be such that the real source could be a factor of 2 higher or a factor of 4 lower.

3.5.5.3 Nuclear Data - cross sections The cross sections used in DTFX and PATH are adequate for shield engineering, but probably cannot produce better than $\pm$ 50% accuracy through a thick shield. Use of the adjusted Mo cross sections in lieu of Zr cross sections could result in an uncertainty of $\pm$ 25%.

3.5.5.4 Nuclear Data - spectra. The accuracy of the neutron spectrum is not believed to have any major impact on the attenuation or dose rate calculations.

3.5.5.5 Nuclear Data - buildup factors. Buildup factors based on older dose conversion factors are known to be inaccurate (about $\pm$ 50%, depending on material, number of mfp's, and gamma energy). Furthermore, buildup factors for complex multi-media configurations which contain short circuit paths are

known to be erroneous. At the side of the FSV-1 cask, with its simple geometry, buildup is assumed to be accurate within a factor of 1.5; at the ends of the cask a factor of 2 should be allowed for.

3.5.5.6 Nuclear Data - dose conversion factors.
Since corrections were made in the analysis for the latest ANS Standard 6.1.1 dose conversion factors, these data are not thought to be a major source of error.

3.5.5.7 Geometry.
Treating the cask as infinitely long instead of a finite length leads to an overestimate of the side neutron dose rates, perhaps by 25%.

The neutron correction factors applied to the ends of the cask were derived from Rockwell (to correct from an infinite slab geometry to a disc source). Again, it is doubtful that the correction has better than $\pm$ 50% accuracy.

The effects of homogenizing the assembly, converting from a square to circular cross section, and simplifying the end hardware are all believed to be negligible.

3.5.5.8 Compositions.
Uncertainties in the compositions of the materials used in these analyses arise from the lack of obtainable information. For example, the detailed compositions of the fuel assembly, stainless steel and neutron shielding are not exactly known. Substitutions have been made such as assuming that stainless steel and cast iron are 100% iron. The uncertainty involved in this is negligible for gamma calculations, but an error of $\pm$ 20% is possible for neutron calculations.

Uncertainties in the densities of the materials are due to composition generalizations used in the calculations. These generalizations are introduced with methods such as homogenizing the fuel assemblies while ignoring their steel support structure. Such generalizations should produce conservative results.

3.5.5.9 Calculational Method.
Uncertainties in the method of calculation are inherent in the computer codes used in this analysis. The uncertainties in DTFX arise from infinite geometry consideration along with quadrature (S_{16}, S_8) considerations. Errors inherent to PATH include the material, single layer buildup factors (factor of 2 on the dose rate) and the breakup

of the source into a finite number of points (about $\pm$ 10% - depending on the geometry).

3.5.5.10 Overall Uncertainty. Combining all of the above systematic and random uncertainties into a single uncertainty factor can be approximated by first separating the uncertainties into the two categories:

Random Uncertainties	Systematic Uncertanties
gamma source terms	Ag110m
neutron source terms	(α, η) conversion
cross sections - general	geometry
cross section - Mo	
buildup factors	
composition	
calculational methods	

The random plus uncertainty for the neutron dose rate is:

$$\sqrt{(100\%)^2 + (50\%)^2 + (25\%)^2 + (20\%)^2 + (10\%)^2} = +117\%$$

Thus, the neutron dose rate could be a factor of approximately 2.2 higher or lower due to random uncertainties. The neutron dose rate could be a factor of approximately 2.7 lower due to systematic uncertainties, and, the combined uncertainty for the neutron dose rate is approximately a factor of 1.0 higher and 3.5 lower.

The random plus uncertainty for the gamma dose rate is:

$$\sqrt{(50\%)^2 + (50\%)^2 + (50\%)^2 + (10\%)^2} = + 87\%$$

Thus, the gamma dose rate due to random uncertainties is approximately a factor of 1.87 higher or lower. The gamma dose rate correction factor for systematic undertainties is a factor of 1.25 lower (for Ag 110m).

Thus the combined uncertainty for the gamma dose rate is approximately a factor of 1.5 higher and 2.3 lower.

Taking the case where 25% of the total dose rate is neutrons and 75% is gammas, the overall approximate uncertainty factors for the dose rate are: 1.4 higher and 2.5 lower, i.e., if the calculated dose rate 6 feet from the vehicle is 10 mrem/hr , the actual dose rate could be as high as 14 mrem/hr or as low as 4 mrem/hr.

3.6 DISCUSSION

The FSV-1 fuel shipping cask has been shown to be adequate without additional shielding to carry one PWR spent fuel element which has experienced 21,000 MWD/MTHM burnup and 2 yr. decay. The external dose rate 6 ft. from the edge of the transport vehicle will be 10 mrem/hr. or less. This dose rate is comprised of 8 mrem/hr. gamma radiation and 2 mrem/hr. neutrons.

The uncertainties in this study are on the conservative side. If:
1. The SANDIA-ORIGEN (α, n) conversion is reduced by a factor of 2, the neutron dose rate will be reduced by 15%.
2. The SANDIA-ORIGEN 2.75-MeV gamma group zeroed out, the gamma dose rate will be reduced by 25%.

Or the maximum permissible burnup can be allowed to rise to $\sim$25,000 MWD/MTHM for a dose rate constraint of 10 mrem/hr.

Because of a higher neutron source strength, BWR fuel elements are more difficult to accommodate in the FSV-1 cask without additional shielding. Five years decay is needed to carry three 19,000 MWD/MTHM BWR fuel elements and meet the 10 mrem/hr. constraint. However, the 2.75-MeV gamma correction is somewhat larger for BWR fuel, so the FSV-1 cask might potentially handle 25,000-MWD/MTHM BWR fuel with 5-yr. decay.

Addition of shielding to the FSV-1 cask greatly enhances the options for carrying LWR spent fuel. The addition of:
1. 6-in. of neutron shielding and 1 in. of gamma shielding, or
2. 3-in. of neutron shielding and 2 in. of gamma shielding.

permits carrying the highest burnup LWR fuel after only one year decay.

It is concluded that from a shielding standpoint the FSV-1 cask has considerably versitility for the transport of LWR spent fuel.

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3.7 REFERENCES

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4.0 WEIGHT STUDY

4.1 SYSTEM WEIGHT EVALUATION FOR THE FSV-1 CASK, MODIFIED TO TRANSPORT LWR FUEL

The FSV-1 Cask, Unit number 1 and 2 with its transport semi-trailer and tractor was designed to operate as a legal system throughout the United States, i.e.; at a Gross Vehicle Weight (GVW) of 73,280 pounds or less. The weight distribution is such that the limit of 32,000 pounds for a tandem axle group is not exceeded. These two units are presently owned and operated by Public Service Company of Colorado.

When FSV-1 Cask Unit number 3, presently owned and operated by General Atomic Company was fabricated, the extendable trailers used with Units 1 and 2 were not available. In order to satisfy the unique requirements of the Fort St. Vrain Nuclear Plant, using a fixed length trailer, it was necessary to mount the cask well to the rear of the trailer. This shift in location, increased the load on the rear axle group to approximately 42,000 pounds and therefore a 3 axle trailer was specified and obtained. This configuration was designed to operate as a legal weight system in those states necessary to transport spent fuel from the Fort St. Vrain reactor in Colorado to the storage facility in Idaho. Individual states have different weight limits for 3 axle groups and thus overweight permits would be required to operate this system in some states.

Four different system weights for the model FSV-1 cask as modified for the transport of canisterized Light Water Reactor (LWR) spent fuel assemblies were developed as part of this evaluation. The system weight is the total weight of the shipping package, the contents, the tiedowns, the semi-trailer and the tractor, i.e., the GVW. The weight of the various components of the FSV-1 CAsk which must be modified for the transport of LWR fuel assemblies are shown in Table 4.1-1. The four cases considered are:

Case 1.   no additional shielding - one PWR fuel assembly

Case 2.   no additional shielding - three BWR fuel assemblies

Case 3.   added neutron shielding - one PWR fuel assembly

Case 4.   added neutron shielding - three BWR fuel assemblies.

The system weight for Case 1 is nearly the same as the operating system weight for the FSV-1 cask configured for the transport of HTGR fuel elements, and thus could operate as a legal weight shipment in some states and overweight in others, as presently mounted on the 3 axle trailer.

Since the expected GVW of 72,920 pounds is just slightly below the 73,280 pound GVW that is legal in all states, it appears likely that with the cask located in the optimum position on a new tandem axle trailer, the actual GVW would be less than 73,280 pounds and the tandem axle loading less than 32,000 pounds.

With the addition of the neutron shielding and the additional weight of three BWR fuel assemblies and their canister it will not be possible to operate at a system weight of 73,280 pounds or less. Forty states now have GVW limits of 80,000 pounds or more, however, there are some other limitations that suggest that the practical limit for a tandem axle semi-trailer with a three axle tractor is about 78,000 pounds. About half of the states limit the load on a tandem axle group to 34,000 pounds and a practical limit for the steering axle of the tractor is about 10,000, thus for the 5 axle tractor and semi-trailer combination under consideration, 78,000 pounds is a practical limit. In states that use "bridge formulas" the distance between the steering axle and the rear axle of the trailer is used to limit the allowable GVW. As an example in the state of California, a 5 axle tractor and semi-trailer combi-nation must measure 48 feet to allow operations at 78,000 pounds.

The maximum flexibility for shipping LWR canisterized fuel assemblies can be achieved by mounting the modified FSV-1 Cask in the optimum location on a new tandem axle trailer. With this configuration and a light weight tractor, the shipping system should operate at 73,280 pounds or less for Case 1 and at 78,000 pounds or less for Cases 2,3,and 4, with the axle loadings within the allowable limits in all states.

WEIGHT EVALUATION FOR THE MODEL FSV-1 CASK
MODIFIED TO SHIP LWR FUEL

	WITHOUT ANY ADDITIONAL SHIELDING		WITH 3" BORO-SILICONE NEUTRON SHIELDING	
	ONE PWR FUEL ASSEMBLY	THREE BWR FUEL ASSEMBLY	ONE PWR FUEL ASSEMBLY	THREE BWR FUEL ASSEMBLY
CASK BODY	40,712	40,712	40,712	40,712
*OUTER CLOSURE	1,035	1,035	1,035	1,035
*INNER CLOSURE	1,368	1,368	1,368	1,368
*IMPACT LIMITER	1,058	1,058	1,058	1,058
FUEL CANISTER-PWR	498	-	498	-
FUEL CANISTER-BWR	-	1,310	-	1,310
FUEL ASSEMBLY-ONE PWR	1,500	-	1,500	-
FUEL ASSEMBLY-THREE BWR	-	2,250	-	2,250
*SPACER-PWR CANISTER	659		659	
*SPACER-BWR CANISTER	-	520	-	520
*NEUTRON SHIELDING			3.530	3,530
SUBTOTAL	46,830	48,253	50,360	51,783
FRONT TIE DOWN	370	370	370	370
REAR TIE DOWN	1,120	1,120	1,120	1,120
*PERSONNEL BARRIER	600	600	600	600
*TRAILER	10,000	10,000	10,000	10,000
TRACTOR	14,000	14,000	14,000	14,000
TOTAL GVW	72,920	74,343	76,450	77,873

* MODIFIED COMPONENTS   Table 4.1-1

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4.2 IMPACT OF OVERWEIGHT TRUCK SHIPMENTS

The 1979 summary of State Size and Weight Limits prepared by the
Truck Trailer Manufacturers Association was used as a basis for evalua-
ting the impact of operating the General Atomic Company, model FSV-1
cask system at a Gross Vehicle Weight (GVW) in excess of 73,280 pounds.
At the present time only 10 states and the District of Columbia limit
GVW's to 73,280 pounds (73,000 pounds for Connecticut). (See Table 4.2-1.)
Due to the location of these states, they effectively block all vehicles
from operating in excess of 73,280 pounds, across the middle of the U.S. and
restrict operations on the East Coast. (See Fig. 4.2-1.) Permits to operate
vehicles in excess of 73,280 pounds are available In each of these 10
states and the District of Columbia, however, the limitations for the
overweight operations vary from state to state. Some states restrict
vehicles operating with overweight permits to travel during daylight
hours only. Some states require several days to process an application
for a permit to operate an overweight vehicle.

Operating overweight vehicles with permits as required by the various
states will require more attention to the planning and scheduling of
shipments, however, General Atomic Company was involved with Philadelphia
Electric Company and Tri-State Motor Transit Company during the shipment
of spent fuel from Peach Bottom 1 located near Delta, Pennsylvania to
the storage facility near Idaho Falls, Idaho. About 90 shipments were
made using the model PB-1 Cask System operating at a GVW of over 100,000
pounds with only minor scheduling problems.

The GVW limits for the various states involve some rather complex
regulations which are continually changing, are sometimes restricted
to certain highways and may be restricted during certain seasons.
Table 4.2-1 should be used for a reference only and the individual
states contacted for all specific shipping operations.

GROSS VEHICLE WEIGHT LIMITS

STATES WITH GVW LIMIT OF 80,000 LB. OR MORE.
(79,800 LB. FOR VIRGINIA AND NORTH CAROLINA)

STATES WITH GVW LIMIT OF 73,280 LB.
(73,000 LB. FOR CONN.)
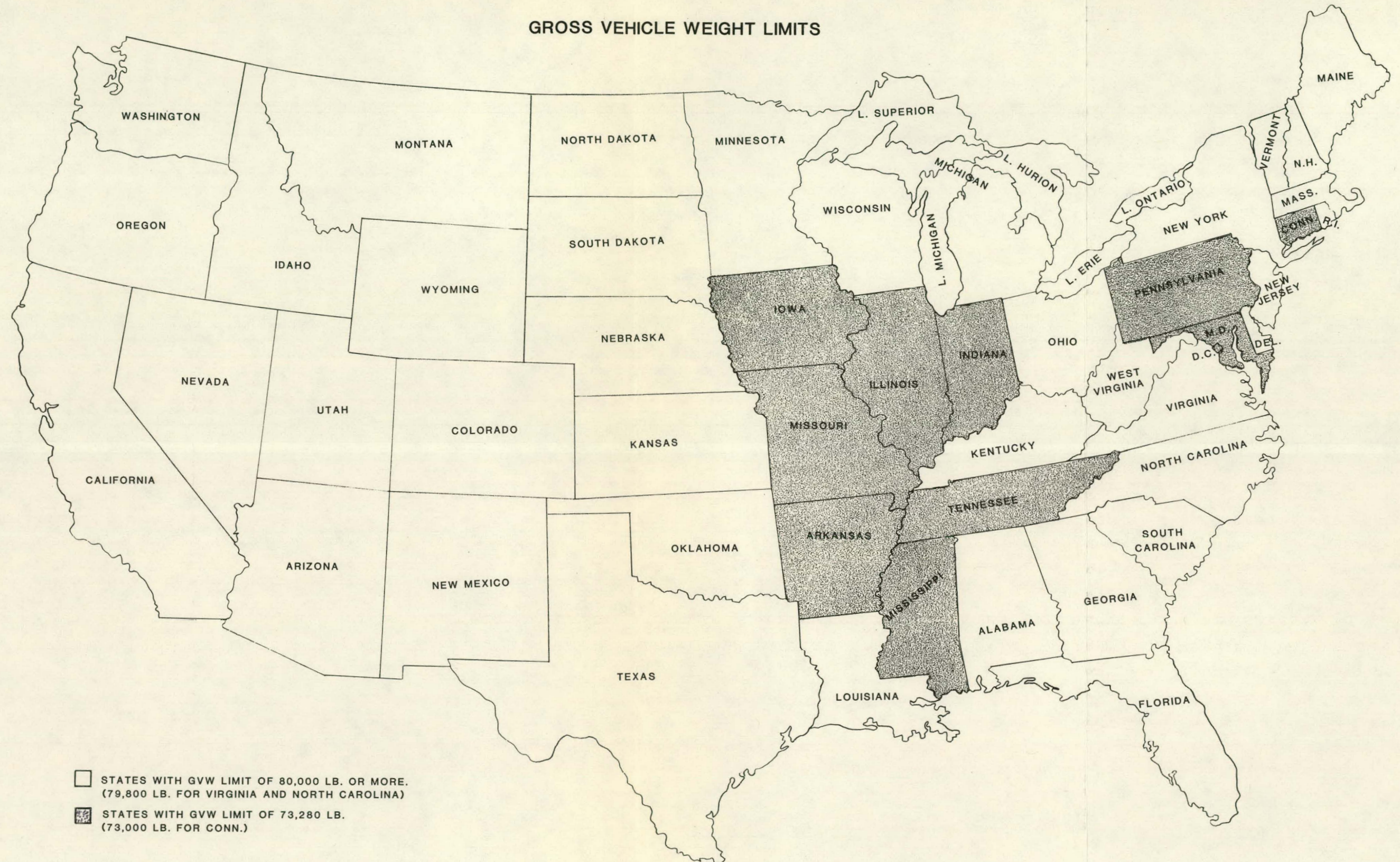
Figure 4.2-1

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STATE WEIGHT LIMITS*

STATE	TANDEM AXLE		GROSS VEHICLE WEIGHT	
	STATE HIGHWAYS	INTERSTATE HIGHWAYS	STATE HIGHWAYS	INTERSTATE HIGHWAYS
Alabama	40,000	36,000	92,400	80,000
Alaska	34,000	NO INTERSTATE	109,000	NO INTERSTATE
Arizona	34,000	34,000	80,000	80,000
Arkansas	32,000	32,000	73,280[a]	73,280
California	34,000	34,000	80,000	80,000
Colorado	36,000	36,000	85,000	80,000
Connecticut	36,000	36,000	73,000	73,000
Delaware	40,000	34,000	80,000	80,000
District of Columbia	37,000	37,000	73,280	73,280
Florida	40,000	40,000	80,000	80,000
Georgia	36,000	36,000	80,000	80,000
Hawaii	34,000	34,000	88,880	80,800
Idaho	34,000	34,000	105,500	80,000
Illinois	32,000	32,000	73,280	73,280
Indiana	32,000	32,000	73,280	73,280
Iowa	32,000	32,000	73,280	73,280
Kansas	34,000	34,000	85,500	80,000
Kentucky	35,700[a]	34,000	82,000[a]	80,000
Louisiana	34,000	34,000	88,000	83,400
Maine	38,000	34,000	80,000	80,000
Maryland	40,000	40,000	73,280	73,280
Massachusetts	36,000	36,000	80,000	80,000
Michigan	34,000[a,b]	34,000	80,000	80,000
Minnesota	34,000[a]	34,000	80,000[a]	80,000
Mississippi	32,000[a]	32,000	73,280[a]	73,280
Missouri	32,000	32,000	73,280[a]	73,280
Montana	32,000	32,000	105,500[c]	105,500[c]
Nebraska	34,000	32,000	95,000	95,000[c]
Nevada	34,000	34,000	129,000[c]	80,000
New Hampshire	36,000	36,000	80,000	80,000
New Jersey	34,000	34,000	80,000	80,000
New Mexico	34,320	34,320	86,400	86,400
New York	36,000	36,000	80,000	80,000
North Carolina	36,000	36,000	79,800	79,800
North Dakota	34,000	34,000	105,500[a]	80,000
Ohio	34,000	34,000	80,000	80,000
Oklahoma	34,000	34,000	90,000	80,000
Oregon	34,000	34,000	80,000	80,000
Pennsylvania	36,000	36,000	73,280	73,280
Rhode Island	36,000	36,000	80,000	80,000
South Carolina	36,000	32,000	80,600	80,000
South Dakota	34,000	34,000	95,000	80,000

a. on designated highways
b. during designated seasons
c. with long term permit

Table 4.2-1

| STATE | TANDEM AXLE | | GROSS VEHICLE WEIGHT | |
	STATE HIGHWAYS	INTERSTATE HIGHWAYS	STATE HIGHWAYS	INTERSTATE HIGHWAYS
Tennessee	32,000	32,000	73,280	73,280
Texas	34,000	34,000	80,000	80,000
Utah	34,000	34,000	80,000	80,000
Vermont	37,800	36,000	80,000	80,000
Virginia	34,000	34,000	79,800	79,800
Washington	34,000	34,000	80,000	80,000
West Virginia	34,000	34,000	80,000[a]	80,000
Wisconsin	34,000	34,000	80,000	80,000
Wyoming	36,000	36,000	101,000	80,000

a. on designated highways
b. during designated seasons
c. with long term permit

STATE SIZE LIMITS

The most restrictive limits for a 5 axle, tractor and semi-trailer combination are:

    Height - 13 feet, 6 inches

    Width - 96 inches

    Length - 55 feet

*Reference: 1979 Chart published by Truck Trailer Manufacturers Association

Table 4.2-1 cont'd.

4.3 DISCUSSION

The major impact of operating with overweight permits, will be
some loss of flexibility of scheduling. When compared to the impact
of satisfying the requirement of the U.S. Nuclear Regulatory Commission
in regard to a safeguarded shipment of radioactive materials i.e.: approved
route, contacts with law enforcement agencies, scheduled reporting, con-
tingency plans, and etc., the impact of operating with overweight permits
will be minimal.

The FSV-1 Cask as modified for the transport of LWR fuels can be
operated as a legalweight system (73,280 pound GVW) for shipments of PWR
fuel assemblies that do not require the addition of neutron shielding.
The qualifying PWR fuel assemblies can be selected by burnup and decay
time. Note that due to the greater weight of the three BWR fuel elements
with that canister and spacer, the system weight will be greater than
73,280 GVW. The added capability of the modified FSV-1 cask with neutron
shielding to carry a wider spectrum of LWR irradiated fuel assemblies
should justify the loss of schedule flexibility required to operate with
overweight permits. The Boro-Silicone neutron shielding will be designed
such that the segments can be removed and should be removed to allow
legal weight operations when possible.

5. THERMAL STUDY

A TAC2D two-dimensional (radial-axial) thermal model of the upper one-half of the cask was developed to calculate interior temperatures. The model is illustrated in Figure 5-1 and it is similar to the TAC2D model used in the design report (GADR-55, Addendum1) for the FSV-1 fuel shipping cask. Property values were also taken from the reference report, except for the Boro-Silicone.[a] Nominal gaps as given in Figure 5-1 were used between internal components.[b] Conduction and radiation heat transfer occur across the gaps. Natural convection and radiation were modeled for the cavities between the cask inner barrel and the fuel canister. The fuel canister was assumed to be supported by four circular plates each 1 inch thick. The fuel region of the TAC2D model is not a true simulation of a LWR fuel bundle. Thus, only the TAC2D temperatures up to and including the fuel canister are valid. The plywood impact limiter was not included in the TAC2D model in order to save nodal points. Instead, the cask surface which is covered by the impact limiter was assumed to be adiabatic in the TAC2D model. The center plane was also assumed adiabatic. Figures 5-2 and 5-3 give the TAC2D steady state temperatures for the interior of the two cask configurations.

Figure 5-4 is a plot of the afterheat in kilowatts for PWR and BWR fuel assemblies vs. the decay time in years for various fuel burnup. The curves are for a single fuel assembly, therefore it is necessary to multiply the values obtained for BWR fuels by 3 since the BWR canister contains 3 fuel assemblies.

The allowable fuel heat load is sensitive to the exterior cask temperature. An increase in the exterior temperature of $10^{\circ}F$ (from $180^{\circ}F$ to $190^{\circ}F$) allows the fuel heat load to increase approximately 50%. At these exterior temperatures the fuel heat load is only about 15-25% of the total surface heat transfer. Since the solar heat load remains fixed with exterior temperature, increasing the exterior temperature by a small amount enables the fuel heat load to be increased significantly. As the exterior temperature increases above $200^{\circ}F$ the fuel heat load becomes more comparable to the solar heat load. Small increases in the exterior temperature no longer

produce as dramatic increases in the allowable fuel heat load. Since
the temperature difference between an interior point and a surface point
is proportional to the fuel heat load, this temperature difference and
the cask interior temperatures are also sensitive to slight changes in
the cask exterior temperature.

(a) Boro-Silicone is a product of Reactor Experiments, Inc. (see
     Appendix D for material properties).

(b) The same model was used for the modified FSV-1 cask with the neutron
     shield except that a .060 gap and the 3 inch neutron shield was
     added external to the cask outer barrel.
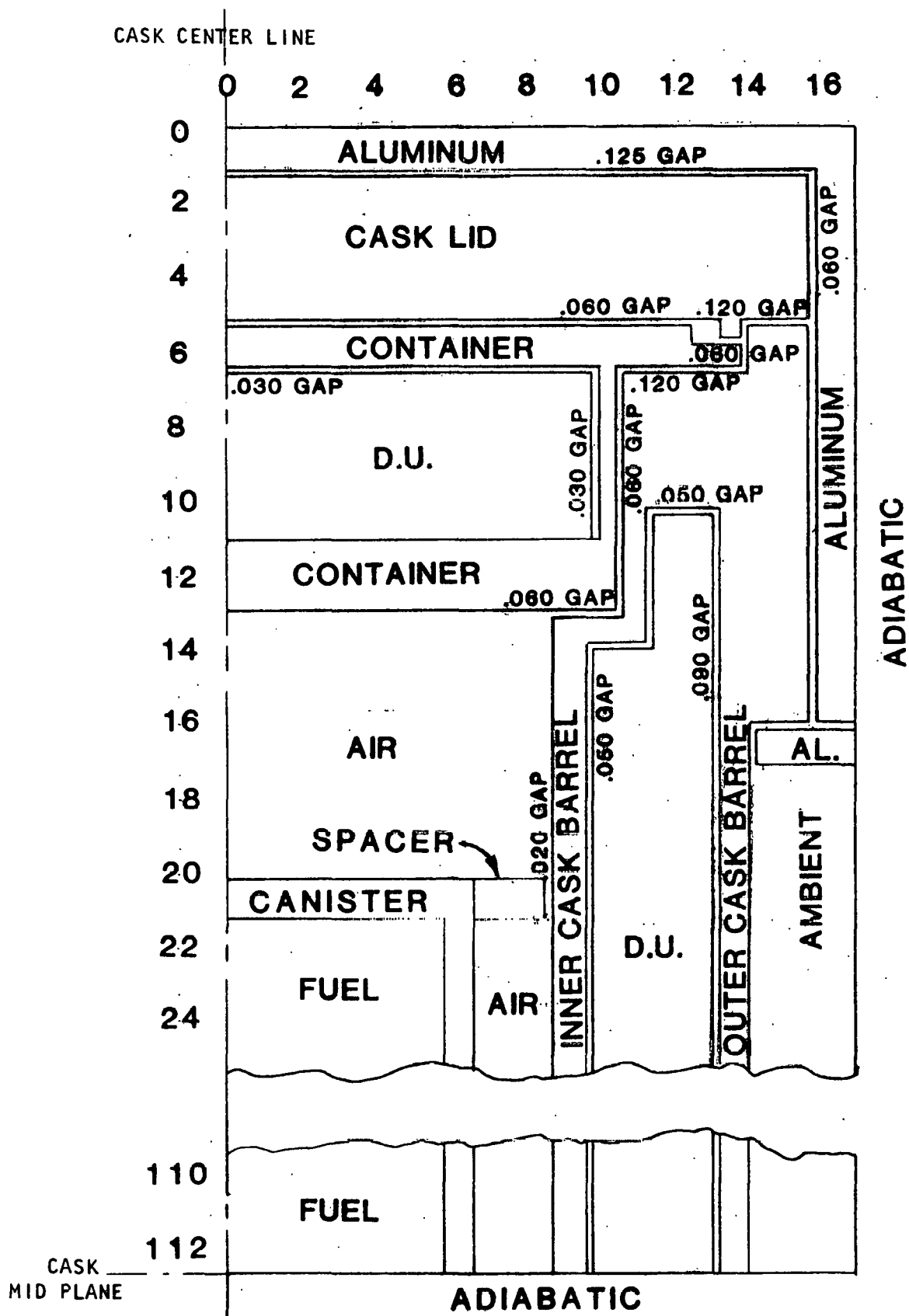
TAC2D two-dimensional thermal model
of upper half of FSV-1 cask (modified) -
without neutron shield
Figure 5-1

124

TEMPERATURES   (F)
THE RADIAL(I) DIRECTION IS HORIZONTAL
THE AXIAL (J) DIRECTION IS VERTICAL

J	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
34																			
33	291	291	290	290	264	226	226	198	190	190	190	190	190	190	180	179	130	130	130
32	290	290	289	289	263	226	226	198	190	190	190	190	190	190	180	180	130	130	130
31	288	288	288	287	262	226	226	198			90	190	190	190	180	180	130	130	130
30	286	286	285	285	255	226	226	201	Inner Cask		91	191	191	180	180	130	130	130
29	285	285	284	283	236	233	228	209	Barrel		91	191	191	180	180	130	130	130
28	285		284	254	222	222	200			91	190	190	180	180	130	130	130
27	286	Fuel	285	259	222	222	197	190	190	189	189	189	189	179	179		
26	285	285	285	284	259	222	222	196	189	189	189	189	188	188	179	179	Ambient
25	282	282	282	281	257	222	222	196	188	188		188	188	188	179	178	
24	278	278	277	277	253	222	222	196	188	188	DU	188	188	188	179	178	130	130	130
23	274	274	273	273	247	222	222	196	188	188		188	188	187	179	179	130	130	130
22	272	272	271	270	240	222	222	196	188	188	188	188	187	187	179	179	130	130	130
21	271	271	270	259	234	222	222	197	188	188	188	187	187	187	179	179	130	130	130
20	237	236	234	232	229	225	219	197	188	187	187				179	179	130	130	130
19	201	201	201	201	201	201	201	195	187	187	187	Outer Cask		180	180	130	130	130
18	201	201	201	201	201	201	201	192	187	187	187	Barrel		181	180	130	130	130
17	201	201		201	201	201	190	187	187	187		181	181	182	182	182
16	201	201	Air	201	201	201	189	187	187	187	187	187	187	182	182	182	182	183
15	201	201			201	201	201	188	187	187	187	187	186	186	182	182	182	182	183
14	201	201	201	201	201	201	201	187	187	187	187	186	186	186	183	183	182	182	183
13	186	186	186	186	186	186	186	186	186	186	186	186	186	186	183	183	183	183	183
12	186	186	186	186	186	186	186	186	186	186	186	186	186	186	183	183	183	183	183
11	186	186	186	186	186	186	186	186	186	185	185	185	186	186	183	183	183	183	183
10	186	186	186	186	186	186	186	186	185	184		184	184	184	184	183	183	183
9	186	186	186	186	DU	186	186	186	185	184	Seal	184	184	184	184	183	183	183
8	186	186	186	186	186	186	185	185	185	185	184	4	184	184	184	183
7	185	185	185	185	185	185	185	184	184	184	184	184	184	184	184	Seal 4	183
6	185	185	185	185	185	185	184	184	184	184	184	184	184	184	4	183
5	184	184	184	184	184	184			4	184	184	184	184	184	184	184	184	183
4	184	184	184	184	184	184	Cask Lid 4	184	184	184	184	184	184	184	184	183
3	184	184	184	184	184	184		4	184	184	184	184	184	184	184	184	183
2	183	183	183	183	183	183	183	183	183	183	183	183	183	183	183	183	183	183	183
1																			

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21

TAC2D results for the cask with no neutron shielding

Figure 5-2

TEMPERATURES   (F)
THE RADIAL(I) DIRECTION IS HORIZONTAL
THE AXIAL (J) DIRECTION IS VERTICAL

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
34	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●
33	●	310	309	309	308	281	240	240	211	202	202	202	202	202	202	191	191	183	182	179	130
32	●	309	308	308	307	280	240	240	211	203	203	202	202	202	202	191	191	183	182	179	130
31	●	307	307	306	306	278	240	240	211	203	203	203	202	203	202	191	191	183	182	179	130
30	●	305	304	304	303	272	240	240	214	204	204	203	203	203	203	191	191	183	182	179	130
29	●	304	303	303	302	252	248	243	223	Inner Cask			203	203	203	191	191	183	182	179	130
28	●	304	303	302	271	237	237	213	Barrel			203	203	203	191	191	183	182	179	130	
27	●	305	Fuel	304	303	276	237	237	210	202	202	201	191	190	182	181	179	130			
26	●	304	304	303	303	276	237	237	209	201	201	201	201	201	201	190	190	182	181	178	130
25	●	301	301	300	300	274	237	237	209	201	201	201	201	201	201	190	190	Boro-		9	130
24	●	296	296	296	295	270	237	237	209	201	201	201	201	201	201	190	190	Silicone		9	130
23	●	292	292	291	291	263	237	237	209	201	201	201	201	201	201	191	191			9	130	Ambient
22	●	290	290	289	288	256	237	237	210	201	201	DU	201	201	201	192	192	184	183	181	130
21	●	289	289	288	287	249	237	237	211	201	201	201	201	201	201	192	192	185	183	181	130
20	●	253	252	251	248	245	240	234	212	201	201	201	201	201	201	193	192	185	184	182	130
19	●	216	216	216	216	216	216	216	209	201	201	201	Outer Cask			193	193	186	185	183	130
18	●	216	216	216	216	216	216	216	206	201	201	201	Barrel			194	194	187	186	184	130
17	●	216	216	216	216	216	216	216	204	201	201	201				195	195	195	195	195	130
16	●	216	216	216	Air	216	216	216	203	201	200	200	200	200	200	195	195	195	195	197	
15	●	216	216	216	216	216	216	216	202	200	200	200	200	200	200	196	196	196	196	197	
14	●	216	216	216	216	216	216	216	202	201	201	201	201	200	200	196	196	196	196	197	
13	●	201	201	201	201	201	201	200	200	200	200	200	200	200	200	197	196	196	196	197	
12	●	201	201	201	201	200	200	200	200	200	200	200	200	200	200	197	197	197	197	197	
11	●	200	200	200	200	200	200	200	200	200	200	199	199	200	200	197	197	197	197	197	
10	●	200	200	200	200	200	200	200	200	200	200	199	199	198	198	198	198	197	197	197	
9	●	200	200	200	200	DU	200	200	200	200	199	198	198	198	198	198	198	197	197		
8	●	200	200	200	200	200	200	200	200	199	198	Seal	198	198	198	198	198	197	197		
7	●	199	199	199	199	199	199	199	199	199	199	199	198	198	198	198	198	198	197		
6	●	199	199	199	199	199	199	199	199	199	199	199	198	198	198	198	Seal	198	197		
5	●	198	198	198	198	198	198	198	198	198	198	198	198	198	198	198	198	198	197		
4	●	198	198	198	198	198	198	Cask Lid	198	198	198	198	198	198	198	198	198	198	197		
3	●	198	198	198	198	198	198	198	198	198	198	198	198	198	198	198	198	197			
2	●	197	197	197	197	197	197	197	197	197	197	197	197	197	197	197	197	197	197	197	
1	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●

5.   TAC2D results for the cask with 3 in. of external
boro-silicone shielding

Figure 5-3

AFTERHEAT (KW) VS. DECAY TIME (YEARS)
FOR VARIOUS BURNUPS (MWD/MTHM)

Figure 5-4

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6.0 DRAWINGS

OUTER CLOSURE NOT
SHOWN THIS AREA

SEAL TEST PORT
(OUTER CLOSURE SEAL)

8° 10' 54"
REF

15° REF

INNER CLOSURE SEAL
TEST AND CAVITY
SAMPLING PORT

CAVITY SAMPLING
VALVE

EXISTING 1 1/4-7 UNC-
24 EQ. SP. ON 25.25
DIA. B.C. (PLUG WITH
SOCKET HEAD SET
SCREW)

NEW 7/8 DIA BOLT
44 EQ SP ON
29.19 DIA B.C.

REMOVE 1.00 FROM
EXISTING CASK

INSTALL HELICOIL
INSERT OR EQUIV.

NEW 1/2 DIA BOLT
24 EQ SP ON 25.25
DIA B.C.

SCALE 1/1

4.75

17.00 DIA

.125

1.25

.120

.38

.06

4.50

D.U.

2.00

.045

D.U.

D.U.

MODIFIED CLOSURE DETAILS

FSV-1 SPENT
FUEL CASK
MODIFICATION

GENERAL ATOMIC
COMPANY
SAN DIEGO, CALIFORNIA

SHEET 5

THREE BWR FUEL ASSEMBLIES
WITH NEUTRON SHIELDING

FSV-1 SPENT
FUEL CASK
MODIFICATION

GENERAL ATOMIC
COMPANY
SAN DIEGO, CALIFORNIA

THREE BWR FUEL ASSEMBLIES
NO NEUTRON SHIELDING

Labels on drawing:
- FRONT SUPPORT UPPER HALF
- SILICONE RUBBER O-RING
- CONCENTRIC SILICONE RUBBER O-RINGS
- LAMINATED PLYWOOD IMPACT LIMITER
- REAR SUPPORT ATTACHMENT BOLTS(4)
- METAL O-RING
- SILICONE RUBBER O-RING
- REAR SUPPORT
- FRONT SUPPORT LOWER HALF
- BWR FUEL ASSY CANISTER 169.69 X 16.00 O.D.
- D. U.

Dimensions:
- 211.63
- 231.39
- 46.75 DIA
- 36.12 DIA
- 31.00 DIA
- 28.00 DIA
- 17.76 DIA
- 188.0
- 21.14
- 34.64
- 7.50
- .25
- 2.00
- 1.25
- 4.75
- 4.50
- .75
- .045
- .125
- .91
- .57
- 3.50
- 2.00
- 6.94

211.63

FRONT SUPPORT
UPPER HALF

SILICONE
RUBBER
O-RING

CONCENTRIC
SILICONE
RUBBER
O-RINGS

46.75
DIA

38.12
DIA

31.00
DIA

.50

.25

2.00

D. U.

185.0

17.76 DIA

PWR FUEL ASSY CANISTER   169.89 X
12.75 O.D.

1.25

4.75   4.50

LAMINATED
PLYWOOD
IMPACT
LIMITER

.75   .045

17.85 TYP 9 PLACES

D. U.

.125

FRONT SUPPORT
LOWER HALF

21.14

34.64

231.39

ONE PWR FUEL ASSEMBLY
WITH NEUTRON SHIELDING

.91

.67   3.50

REAR SUPPORT
ATTACHMENT
BOLTS(4)

28.00 DIA

METAL O-RING

SILICONE
RUBBER
O-RING

2.00

8.94

3.00

REAR SUPPORT

BORO-SILICONE

LIST OF MATERIAL

REQD/ASSEMBLY

FSV-1 SPENT
FUEL CASK
MODIFICATION

GENERAL ATOMIC
COMPANY
SAN DIEGO, CALIFORNIA

REVISED

SHEET 2

ONE PWR FUEL ASSEMBLY
NO NEUTRON SHIELDING

FRONT SUPPORT
UPPER HALF

SILICONE
RUBBER
O-RING

CONCENTRIC
SILICONE
RUBBER
O-RINGS

LAMINATED
PLYWOOD
IMPACT
LIMITER

FRONT SUPPORT
LOWER HALF

REAR SUPPORT
ATTACHMENT
BOLTS(4)

METAL O-RING

SILICONE
RUBBER
O-RING

REAR SUPPORT

PWR FUEL ASSY CANISTER   169.89 X
12.75 O.D.

211.83
188.0
231.39

48.75 DIA
38.12 DIA
31.00 DIA
17.76 DIA
28.00 DIA

.50
.25
.75
1.25
4.75
2.00
4.50
.045
.125
17.05 TYP 9 PLACES
D. U.
D. U.
21.14
34.84

.91
.57
3.50
2.00
8.94

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7.0 APPENDIX

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Table A-1 Fission Product Gamma Source Strengths

FISSION PRODUCTS	PWR-CYCLIC OPERATION-20,000MWD/MTHM	AVERAGE POWER(MW)	1.54956E+01
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.08511E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV	BASIS = ONE FUEL ASSEMBLY	TOTAL BURNUP(MWD)	9.14240E+03

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	4.398E+16	2.685E+12	7.013E+11	3.619E+11	2.558E+11	1.950E+11	1.508E+11	1.169E+11	9.066E+10	7.033E+10	5.456E+10
.63	0.	2.715E+17	3.523E+15	2.053E+15	1.576E+15	1.248E+15	1.024E+15	8.666E+14	7.545E+14	6.728E+14	6.120E+14	5.666E+14
1.10	0.	1.964E+17	1.675E+14	1.230E+14	9.601E+13	7.851E+13	6.645E+13	5.764E+13	5.086E+13	4.543E+13	4.092E+13	3.708E+13
1.55	0.	2.772E+17	1.003E+14	6.117E+13	3.921E+13	2.605E+13	1.779E+13	1.234E+13	8.656E+12	6.111E+12	4.332E+12	3.080E+12
1.99	0.	1.772E+17	9.628E+13	3.957E+13	1.654E+13	6.676E+12	2.867E+12	1.199E+12	5.035E+11	2.123E+11	8.997E+10	3.835E+10
2.38	0.	9.310E+16	3.311E+12	1.667E+12	8.335E+11	4.227E+11	2.125E+11	1.072E+11	5.396E+10	2.717E+10	1.368E+10	6.888E+09
2.75	0.	3.895E+16	2.001E+13	7.416E+12	2.760E+12	1.034E+12	3.910E+11	1.500E+11	5.899E+10	2.425E+10	1.082E+10	5.550E+09
3.25	0.	1.750E+16	2.305E+04	5.829E-05	1.474E-13	3.727E-22	9.423E-31	2.383E-39	6.025E-48	1.523E-56	3.852E-65	9.740E-74
3.70	0.	1.473E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	1.133E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	1.557E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	2.290E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	1.146E+18	3.913E+15	2.317E+15	1.732E+15	1.361E+15	1.111E+15	9.332E+14	8.148E+14	7.247E+14	6.575E+14	6.062E+14

FISSION PRODUCTS	PWR-CYCLIC OPERATION-33,000MWD/MTHM	AVERAGE POWER(MW)	1.70230E+01
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.64316E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV	BASIS = ONE FUEL ASSEMBLY	TOTAL BURNUP(MWD)	1.50654E+04

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	1.168E+17	4.957E+12	1.241E+12	6.559E+11	4.669E+11	3.563E+11	2.756E+11	2.137E+11	1.658E+11	1.286E+11	9.575E+10
.63	0.	7.063E+17	7.846E+15	4.709E+15	3.507E+15	2.707E+15	2.154E+15	1.765E+15	1.488E+15	1.287E+15	1.135E+15	1.025E+15
1.10	0.	5.312E+17	4.120E+14	3.159E+14	2.547E+14	2.131E+14	1.832E+14	1.605E+14	1.425E+14	1.278E+14	1.155E+14	1.046E+14
1.55	0.	7.506E+17	2.221E+14	1.436E+14	9.604E+13	6.577E+13	4.574E+13	3.212E+13	2.269E+13	1.610E+13	1.145E+13	8.154E+12
1.99	0.	4.868E+17	1.321E+14	5.487E+13	2.284E+13	9.534E+12	3.993E+12	1.679E+12	7.096E+11	3.014E+11	1.288E+11	5.546E+10
2.38	0.	2.555E+17	5.592E+12	3.017E+12	1.519E+12	7.649E+11	3.851E+11	1.939E+11	9.764E+10	4.916E+10	2.475E+10	1.246E+10
2.75	0.	1.096E+17	6.758E+13	2.490E+13	9.200E+12	3.411E+12	1.271E+12	4.780E+11	1.824E+11	7.170E+10	2.987E+10	1.391E+10
3.25	0.	4.947E+16	6.240E+04	1.578E-04	3.990E-13	1.009E-21	2.551E-30	6.450E-39	1.631E-47	4.124E-56	1.043E-64	2.637E-73
3.70	0.	3.826E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	2.919E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	4.029E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	6.098E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	3.084E+18	8.451E+15	5.253E+15	3.892E+15	3.000E+15	2.389E+15	1.961E+15	1.654E+15	1.431E+15	1.266E+15	1.142E+15

Table A-1 Fission Product Gamma Source Strengths cont'd.

FISSION PRODUCTS	PWR-CYCLIC OPERATION-40,000MWD/MTHM	AVERAGE POWER(MW)	2.06340E+01
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	4.59108E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV BASIS = ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	1.82611E+04

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	1.436E+17	6.386E+12	1.614E+12	8.551E+11	6.090E+11	4.648E+11	3.595E+11	2.788E+11	2.162E+11	1.677E+11	1.301E+11
.63	0.	8.745E+17	1.055E+16	6.607E+15	4.888E+15	3.736E+15	2.937E+15	2.375E+15	1.974E+15	1.685E+15	1.473E+15	1.317E+15
1.10	0.	6.569E+17	5.748E+14	4.424E+14	3.572E+14	2.990E+14	2.568E+14	2.248E+14	1.995E+14	1.788E+14	1.613E+14	1.464E+14
1.55	0.	9.291E+17	3.147E+14	2.060E+14	1.391E+14	9.578E+13	6.685E+13	4.706E+13	3.330E+13	2.364E+13	1.682E+13	1.199E+13
1.99	0.	6.081E+17	1.625E+14	6.746E+13	2.811E+13	1.175E+13	4.927E+12	2.075E+12	8.785E+11	3.740E+11	1.603E+11	6.917E+10
2.38	0.	3.190E+17	7.897E+12	3.976E+12	2.002E+12	1.008E+12	5.076E+11	2.556E+11	1.287E+11	6.480E+10	3.263E+10	1.643E+10
2.75	0.	1.386E+17	1.053E+14	4.024E+13	1.484E+13	5.490E+12	2.040E+12	7.632E+11	2.892E+11	1.122E+11	4.578E+10	2.006E+10
3.25	0.	6.263E+16	7.726E+04	1.354E-04	9.941E-13	1.249E-21	3.159E-30	7.988E-39	2.020E-47	5.107E-56	1.251E-64	3.265E-73
3.70	0.	4.656E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	3.544E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	4.901E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	7.530E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	3.827E+18	1.172E+16	7.368E+15	5.430E+15	4.149E+15	3.269E+15	2.650E+15	2.208E+15	1.888E+15	1.652E+15	1.476E+15

FISSION PRODUCTS	GE STD BWR 6-4 CYCLE=2.66 W/C HIGH	AVERAGE POWER(MW)	5.65050E+00
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.42695E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV BASIS= ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	6.21555E+03

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	3.372E+16	1.545E+12	4.367E+11	2.479E+11	1.795E+11	1.378E+11	1.066E+11	8.269E+10	6.414E+10	4.976E+10	3.860E+10
.63	0.	2.029E+17	2.762E+15	1.789E+15	1.341E+15	1.040E+15	8.308E+14	6.832E+14	5.775E+14	5.009E+14	4.446E+14	4.025E+14
1.10	0.	1.518E+17	1.632E+14	1.275E+14	1.042E-14	8.811E+13	7.632E+13	6.725E+13	5.998E+13	5.396E+13	4.885E+13	4.442E+13
1.55	0.	2.145E+17	8.059E+13	5.300E+13	3.587E+13	2.475E+13	1.729E+13	1.218E+13	8.622E+12	6.123E+12	4.357E+12	3.104E+12
1.99	0.	1.395E+17	3.858E+13	1.606E+13	6.702E+12	2.807E+12	1.180E+12	4.985E+11	2.117E+11	9.048E+10	3.893E+10	1.688E+10
2.38	0.	7.306E+16	2.104E+12	1.059E+12	5.334E+11	2.686E+11	1.352E+11	6.809E+10	3.429E+10	1.726E+10	8.692E+09	4.377E+09
2.75	0.	3.168E+16	2.663E+13	9.806E+12	3.619E+12	1.340E+12	4.988E+11	1.872E+11	7.133E+10	2.802E+10	1.171E+10	5.515E+09
3.25	0.	1.431E+16	1.777E+04	4.494E-05	1.136E-13	2.873E-22	7.265E-31	1.837E-39	4.645E-48	1.175E-56	2.970E-65	7.510E-74
3.70	0.	1.067E+16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	8.081E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	1.120E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	1.710E+15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	8.831E+17	3.075E+15	1.997E+15	1.492E+15	1.157E+15	9.264E+14	7.635E+14	6.465E+14	5.612E+14	4.979E+14	4.501E+14

Table A-2 Neutron Source Strength

ACTINIDES	PWR-CYCLIC OPERATION-20,000MWD/MTHM	AVERAGE POWER(MW)	1.54956E+01
NUCLIDE LISTING	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.08511E+13
ALPHA N NEUTRON RATE - NEUT/SEC	BASIS = ONE FUEL ASSEMBLY	TOTAL BURNUP(MWD)	9.14240E+03

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
BI211	0.	4.655E-04	1.192E-03	1.869E-03	2.543E-03	3.239E-03	3.948E-03	4.670E-03	5.404E-03	6.150E-03	6.907E-03	7.675E-03
BI212	0.	1.305E-02	3.154E-02	5.105E-02	7.036E-02	8.893E-02	1.030E-01	1.156E-01	1.260E-01	1.344E-01	1.411E-01	1.463E-01
BI213	0.	1.066E-09	5.872E-10	7.876E-10	9.901E-10	1.196E-09	1.406E-09	1.618E-09	1.834E-09	2.053E-09	2.275E-09	2.501E-09
PO210	0.	6.879E-08	2.722E-07	6.958E-07	1.429E-06	2.562E-06	4.179E-06	6.366E-06	9.204E-06	1.277E-05	1.714E-05	2.240E-05
PO211	0.	2.216E-06	5.673E-06	8.957E-06	1.211E-05	1.542E-05	1.680E-05	2.223E-05	2.573E-05	2.928E-05	3.268E-05	3.654E-05
PO212	0.	1.362E+00	3.334E+00	5.413E+00	7.399E+00	9.191E+00	1.075E+01	1.207E+01	1.315E+01	1.403E+01	1.473E+01	1.527E+01
PO213	0.	9.893E-05	5.350E-05	7.171E-05	9.021E-05	1.090E-04	1.281E-04	1.475E-04	1.671E-04	1.871E-04	2.073E-04	2.279E-04
PO214	0.	2.392E-05	5.146E-05	9.094E-05	1.424E-04	2.055E-04	2.812E-04	3.686E-04	4.681E-04	5.796E-04	7.032E-04	6.386E-04
PO215	0.	7.165E-04	1.834E-03	2.864E-03	3.914E-03	4.986E-03	6.077E-03	7.188E-03	8.318E-03	9.466E-03	1.063E-02	1.181E-02
PO216	0.	8.283E-01	2.027E+00	3.291E+00	4.499E+00	5.589E+00	6.536E+00	7.336E+00	7.997E+00	8.532E+00	8.956E+00	9.285E+00
PO218	0.	1.035E-05	2.227E-05	3.936E-05	6.162E-05	8.906E-05	1.217E-04	1.595E-04	2.026E-04	2.508E-04	3.043E-04	3.631E-04
AT217	0.	5.450E-05	2.945E-05	3.948E-05	4.966E-05	6.001E-05	7.051E-05	8.116E-05	9.200E-05	1.030E-04	1.141E-04	1.254E-04
RN219	0.	5.373E-04	1.376E-03	2.148E-03	2.936E-03	3.739E-03	4.558E-03	5.391E-03	6.238E-03	7.099E-03	7.973E-03	8.860E-03
RN220	0.	6.278E-01	1.535E+00	2.491E+00	3.405E+00	4.231E+00	4.947E+00	5.553E+00	6.054E+00	6.458E+00	6.779E+00	7.028E+00
RN222	0.	7.034E-06	1.513E-05	2.674E-05	4.137E-05	6.051E-05	8.269E-05	1.084E-04	1.376E-04	1.704E-04	2.068E-04	2.467E-04
FR221	0.	3.526E-05	1.906E-05	2.554E-05	3.213E-05	3.882E-05	4.562E-05	5.252E-05	5.952E-05	6.663E-05	7.384E-05	8.116E-05
RA223	0.	3.075E-04	7.873E-04	1.229E-03	1.680E-03	2.140E-03	2.609E-03	3.086E-03	3.570E-03	4.063E-03	4.563E-03	5.071E-03
RA224	0.	4.308E-01	1.054E+00	1.712E+00	2.340E+00	2.907E+00	3.399E+00	3.816E+00	4.159E+00	4.437E+00	4.658E+00	4.829E+00
RA226	0.	4.207E-06	9.051E-06	1.599E-05	2.504E-05	3.619E-05	4.946E-05	6.483E-05	8.233E-05	1.019E-04	1.237E-04	1.475E-04
AC225	0.	2.631E-05	1.422E-05	1.936E-05	2.357E-05	2.896E-05	3.404E-05	3.918E-05	4.441E-05	4.971E-05	5.509E-05	6.055E-05
AC227	0.	9.682E-13	2.145E-12	3.347E-12	4.575E-12	5.826E-12	7.101E-12	8.399E-12	9.718E-12	1.106E-11	1.242E-11	1.380E-11
TH227	0.	3.099E-04	7.500E-04	1.171E-03	1.601E-03	2.039E-03	2.485E-03	2.939E-03	3.401E-03	3.871E-03	4.347E-03	4.831E-03
TH228	0.	3.661E-01	8.981E-01	1.457E+00	1.990E+00	2.472E+00	2.891E+00	3.244E+00	3.536E+00	3.772E+00	3.959E+00	4.105E+00
TH229	0.	5.946E-06	6.941E-06	1.193E-05	1.508E-05	1.822E-05	2.140E-05	2.464E-05	2.793E-05	3.126E-05	3.465E-05	3.808E-05
TH230	0.	8.711E-03	1.353E-02	1.736E-02	2.319E-02	2.804E-02	3.290E-02	3.777E-02	4.264E-02	4.753E-02	5.243E-02	5.733E-02
TH232	0.	3.593E-09	6.736E-09	9.378E-09	1.302E-08	1.616E-08	1.931E-08	2.245E-08	2.559E-08	2.874E-08	3.188E-08	3.502E-08
PA231	0.	8.488E-03	8.962E-03	9.417E-03	9.672E-03	1.033E-02	1.078E-02	1.124E-02	1.169E-02	1.215E-02	1.260E-02	1.306E-02
U232	0.	1.596E+00	2.259E+00	2.766E+00	3.154E+00	3.449E+00	3.671E+00	3.836E+00	3.956E+00	4.041E+00	4.099E+00	4.135E+00
U233	0.	2.722E-02	2.766E-02	2.811E-02	2.856E-02	2.901E-02	2.946E-02	2.990E-02	3.035E-02	3.080E-02	3.125E-02	3.170E-02
U234	9.307E+02	5.943E+02	5.955E+02	5.963E+02	5.980E+02	5.992E+02	6.004E+02	6.016E+02	6.028E+02	6.040E+02	6.052E+02	6.063E+02
U235	3.341E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01	1.503E+01
U236	0.	9.693E+01	9.693E+01	9.694E+01	9.694E+01	9.694E+01	9.695E+01	9.695E+01	9.695E+01	9.696E+01	9.696E+01	9.697E+01
U238	1.066E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02	1.070E+02
NP237	0.	9.624E+01	9.765E+01	9.771E+01	9.775E+01	9.731E+01	9.788E+01	9.796E+01	9.805E+01	9.816E+01	9.827E+01	9.839E+01
PU236	0.	1.063E+02	8.353E+01	6.554E+01	5.139E+01	4.029E+01	3.159E+01	2.477E+01	1.942E+01	1.523E+01	1.194E+01	9.364E+00
PU238	0.	7.115E+05	7.334E+05	7.326E+05	7.279E+05	7.225E+05	7.170E+05	7.114E+05	7.059E+05	7.005E+05	6.951E+05	6.897E+05
PU239	0.	2.167E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05	2.181E+05
PU240	0.	2.266E+05	2.266E+05	2.266E+05	2.266E+05	2.266E+05	2.266E+05	2.266E+05	2.266E+05	2.265E+05	2.265E+05	2.265E+05
PU241	0.	2.793E-08	2.664E-08	2.540E-08	2.423E-08	2.310E-08	2.203E-08	2.101E-08	2.004E-08	1.911E-08	1.822E-08	1.738E-08
PU242	0.	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02	2.065E+02
AM241	0.	3.769E+04	1.108E+05	1.804E+05	2.467E+05	3.098E+05	3.698E+05	4.269E+05	4.813E+05	5.330E+05	5.822E+05	6.290E+05
AM243	0.	2.196E+03	2.193E+03	2.193E+03	2.198E+03	2.196E+03	2.196E+03	2.197E+03	2.197E+03	2.197E+03	2.197E+03	2.197E+03
CM242	0.	8.494E+06	1.811E+06	3.358E+05	3.416E+04	2.034E+04	6.023E+03	3.953E+03	3.334E+03	3.191E+03	3.150E+03	3.130E+03
CM243	0.	6.053E+02	5.923E+02	5.796E+02	5.672E+02	5.551E+02	5.432E+02	5.315E+02	5.201E+02	5.090E+02	4.981E+02	4.874E+02
CM244	0.	2.030E+05	1.954E+05	1.881E+05	1.810E+05	1.742E+05	1.677E+05	1.614E+05	1.553E+05	1.495E+05	1.439E+05	1.385E+05
CM245	0.	1.269E+01	1.269E+01	1.269E+01	1.269E+01	1.269E+01	1.269E+01	1.269E+01	1.269E+01	1.269E+01	1.268E+01	1.268E+01
CM246	0.	1.511E+00	1.511E+00	1.511E+00	1.510E+00	1.510E+00	1.510E+00	1.510E+00	1.509E+00	1.509E+00	1.509E+00	1.509E+00
CM247	0.	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06	2.742E-06
CM248	0.	6.686E-04	6.686E-04	6.686E-04	6.686E-04	6.687E-04	6.687E-04	6.687E-04	6.687E-04	6.687E-04	6.687E-04	6.687E-04
CF249	0.	1.203E-05	5.019E-05	6.714E-05	7.462E-05	7.767E-05	7.924E-05	7.976E-05	7.990E-05	7.988E-05	7.978E-05	7.965E-05
CF250	0.	3.666E-04	3.486E-04	3.306E-04	3.135E-04	2.973E-04	2.820E-04	2.674E-04	2.536E-04	2.405E-04	2.281E-04	2.163E-04
CF251	0.	8.779E-07	8.772E-07	8.766E-07	8.759E-07	8.752E-07	8.745E-07	8.739E-07	8.732E-07	8.725E-07	8.718E-07	8.712E-07
CF252	0.	1.060E-03	8.156E-04	6.277E-04	4.830E-04	3.717E-04	2.860E-04	2.201E-04	1.694E-04	1.303E-04	1.003E-04	7.719E-05
CF254	0.	4.860E-07	7.399E-09	1.125E-10	1.857E-12	2.647E-13	2.047E-13	3.191E-13	2.467E-13	3.847E-13	2.974E-13	2.299E-13
ES253	0.	3.679E-06	1.401E-10	7.017E-16	3.959E-21	1.981E-26	9.900E-32	4.945E-37	2.470E-42	1.234E-47	6.164E-53	3.079E-58
TOTAL		1.073E+03	9.894E+06	3.300E+06	1.936E+06	1.638E+06	1.675E+06	1.710E+06	1.752E+06	1.794E+06	1.835E+06	1.873E+06	1.909E+06

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ACTINIDES			PWR-CYCLIC OPERATION-23,500MWD/MTHM				AVERAGE POWER(MW)		1.54956E+01		
NUCLIDE LISTING			1ST SPENT FUEL DECAY UP TO 10 YRS				AVERAGE FLUX(N/CM**2-SEC)		3.08511E+13		
SPONTANEOUS FISSION NEUT RATE-NEUT/SEC		BASIS = ONE FUEL ASSEMBLY					TOTAL BURNUP(MWD)		9.14240E+03		

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
PU236	0.	2.515E+00	1.973E+00	1.551E+00	1.216E+00	9.535E-01	7.476E-01	5.862E-01	4.596E-01	3.604E-01	2.826E-01	2.216E-01
PU238	0.	4.940E+04	5.092E+04	5.036E+04	5.054E+04	5.017E+04	4.978E+04	4.940E+04	4.902E+04	4.864E+04	4.826E+04	4.789E+04
PU240	0.	6.129E+05	6.129E+05	6.128E+05	6.128E+05	6.128E+05	6.128E+05	6.127E+05	6.127E+05	6.127E+05	6.126E+05	6.126E+05
PU242	0.	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04	8.159E+04
PU244	0.	7.824E-10	3.645E-09	6.508E-09	9.371E-09	1.223E-08	1.510E-08	1.796E-08	2.082E-08	2.369E-08	2.655E-08	2.941E-09
CM242	0.	1.729E+07	3.686E+06	7.050E+05	1.713E+05	4.139E+04	1.386E+04	8.043E+03	6.784E+03	6.495E+03	6.410E+03	6.370E+03
CM244	0.	1.185E+07	1.141E+07	1.093E+07	1.057E+07	1.017E+07	9.789E+06	9.421E+06	9.067E+06	8.726E+06	8.399E+06	8.083E+06
CM246	0.	2.367E+04	2.367E+04	2.367E+04	2.366E+04	2.366E+04	2.366E+04	2.365E+04	2.365E+04	2.364E+04	2.364E+04	2.364E+04
CM248	0.	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01	2.892E+01
CM250	0.	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05	1.702E-05
CF252	0.	1.416E+02	1.090E+02	8.385E+01	6.453E+01	4.966E+01	3.821E+01	2.941E+01	2.263E+01	1.741E+01	1.340E+01	1.031E+01
CF254	0.	2.646E-02	4.028E-04	6.126E-06	1.011E-07	1.441E-03	1.114E-08	-1.737E-08	-1.343E-08	2.094E-08	1.619E-08	1.252E-08
TOTAL	0.	2.991E+07	1.586E+07	1.253E+07	1.151E+07	1.098E+07	1.057E+07	1.020E+07	9.841E+06	9.500E+06	9.171E+06	8.855E+06
TOTAL	1.073E+03	3.980E+07	1.316E+07	1.447E+07	1.320E+07	1.266E+07	1.229E+07	1.195E+07	1.164E+07	1.133E+07	1.104E+07	1.076E+07

Table A-2 Neutron Source Strength cont'd.

ACTINIDES			PWR-CYCLIC OPERATION-33,000MWD/MTHM		AVERAGE POWER(MW)	1.70230E+01
NUCLIDE LISTING			1ST SPENT FUEL DECAY UP TO 10 YRS		AVERAGE FLUX(N/CM**2-SEC)	3.64316E+13
ALPHA N NEUTRON RATE - NEUT/SEC			BASIS = ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	1.50654E+04

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
BI211	0.	1.390E-03	2.786E-03	4.019E-03	5.232E-03	6.425E-03	7.599E-03	8.754E-03	9.890E-03	1.101E-02	1.211E-02	1.320E-02
BI212	0.	4.560E-02	1.071E-01	1.765E-01	2.451E-01	3.084E-01	3.643E-01	4.119E-01	4.517E-01	4.840E-01	5.059E-01	5.301E-01
BI213	0.	9.395E-09	1.655E-09	1.856E-09	2.064E-09	2.278E-09	2.500E-09	2.728E-09	2.963E-09	3.205E-09	3.453E-09	3.708E-09
PO210	0.	2.641E-07	6.310E-07	1.204E-06	2.042E-06	3.205E-06	4.755E-06	6.751E-06	9.252E-06	1.231E-05	1.559E-05	2.035E-05
PO211	0.	6.619E-06	1.326E-05	1.913E-05	2.491E-05	3.059E-05	3.618E-05	4.167E-05	4.709E-05	5.242E-05	5.766E-05	6.282E-05
PO212	0.	4.761E+01	1.118E+01	1.843E+01	2.559E+01	3.220E+01	3.803E+01	4.301E+01	4.716E+01	5.054E+01	5.323E+01	5.534E+01
PO213	0.	8.560E-04	1.508E-04	1.691E-04	1.881E-04	2.076E-04	2.278E-04	2.486E-04	2.700E-04	2.920E-04	3.146E-04	3.375E-04
PO214	0.	4.549E-05	7.143E-05	1.058E-04	1.486E-04	2.001E-04	2.603E-04	3.292E-04	4.070E-04	4.937E-04	5.855E-04	6.944E-04
PO215	0.	2.140E-03	4.288E-03	6.186E-03	8.053E-03	9.890E-03	1.170E-02	1.347E-02	1.522E-02	1.695E-02	1.864E-02	2.032E-02
PO216	0.	2.895E+00	6.800E+00	1.121E+01	1.555E+01	1.958E+01	2.312E+01	2.615E+01	2.867E+01	3.073E+01	3.237E+01	3.365E+01
PO218	0.	1.965E-05	3.051E-05	4.578E-05	6.434E-05	8.661E-05	1.126E-04	1.425E-04	1.761E-04	2.137E-04	2.551E-04	3.005E-04
AT217	0.	4.713E-04	8.301E-05	9.310E-05	1.035E-04	1.143E-04	1.254E-04	1.368E-04	1.486E-04	1.607E-04	1.732E-04	1.862E-04
RN219	0.	1.605E-03	3.216E-03	4.640E-03	6.040E-03	7.417E-03	8.772E-03	1.011E-02	1.142E-02	1.271E-02	1.398E-02	1.524E-02
RN220	0.	2.191E+00	5.147E+00	8.482E+00	1.178E+01	1.482E+01	1.750E+01	1.980E+01	2.170E+01	2.326E+01	2.450E+01	2.547E+01
RN222	0.	1.338E-05	2.100E-05	3.111E-05	4.371E-05	5.885E-05	7.654E-05	9.680E-05	1.197E-04	1.452E-04	1.723E-04	2.042E-04
FR221	0.	3.049E-04	5.371E-05	6.024E-05	6.698E-05	7.395E-05	8.113E-05	8.853E-05	9.616E-05	1.040E-04	1.121E-04	1.203E-04
RA223	0.	9.186E-04	1.841E-03	2.655E-03	3.457E-03	4.245E-03	5.021E-03	5.784E-03	6.535E-03	7.274E-03	8.003E-03	8.720E-03
RA224	0.	1.506E+00	3.537E+00	5.828E+00	8.094E+00	1.018E+01	1.203E+01	1.360E+01	1.491E+01	1.683E+01	1.752E+01	1.750E+01
RA226	0.	8.001E-06	1.256E-05	1.861E-05	2.615E-05	3.520E-05	4.578E-05	5.790E-05	7.158E-05	8.684E-05	1.037E-04	1.221E-04
AC225	0.	2.275E-04	4.007E-05	4.494E-05	4.997E-05	5.517E-05	6.053E-05	6.605E-05	7.174E-05	7.759E-05	8.360E-05	8.976E-05
AC227	0.	2.757E-12	5.013E-12	7.230E-12	9.411E-12	1.156E-11	1.367E-11	1.574E-11	1.779E-11	1.980E-11	2.178E-11	2.373E-11
TH227	0.	9.082E-04	1.753E-03	2.530E-03	3.293E-03	4.044E-03	4.783E-03	5.510E-03	6.225E-03	6.930E-03	7.624E-03	8.308E-03
TH228	0.	1.279E+00	3.012E+00	4.960E+00	6.886E+00	8.661E+00	1.023E+01	1.156E+01	1.268E+01	1.359E+01	1.431E+01	1.488E+01
TH229	0.	2.224E-05	2.520E-05	2.826E-05	3.143E-05	3.469E-05	3.806E-05	4.154E-05	4.511E-05	4.879E-05	5.256E-05	5.646E-05
TH230	0.	8.787E-03	1.217E-02	1.559E-02	1.905E-02	2.255E-02	2.608E-02	2.965E-02	3.326E-02	3.690E-02	4.058E-02	4.427E-02
TH232	0.	7.123E-09	1.121E-08	1.529E-08	1.938E-08	2.346E-08	2.754E-08	3.163E-08	3.571E-08	3.980E-08	4.389E-08	4.797E-08
PA231	0.	1.677E-02	1.707E-02	1.731E-02	1.755E-02	1.779E-02	1.803E-02	1.827E-02	1.851E-02	1.875E-02	1.899E-02	1.922E-02
U232	0.	5.027E+00	7.638E+00	9.636E+00	1.117E+01	1.233E+01	1.321E+01	1.387E+01	1.435E+01	1.469E+01	1.493E+01	1.508E+01
U233	0.	2.664E-02	2.758E-02	2.852E-02	2.947E-02	3.042E-02	3.136E-02	3.231E-02	3.326E-02	3.421E-02	3.516E-02	3.611E-02
U234	9.307E+02	4.156E+02	4.202E+02	4.248E+02	4.295E+02	4.341E+02	4.386E+02	4.431E+02	4.476E+02	4.521E+02	4.565E+02	4.605E+02
U235	3.341E+01	7.907E+00	7.907E+00	7.908E+00	7.908E+00	7.908E+00	7.908E+00	7.908E+00	7.908E+00	7.908E+00	7.905E+00	7.905E+00
U236	0.	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02	1.260E+02
U238	1.086E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02	1.058E+02
NP237	0.	2.005E+02	2.063E+02	2.064E+02	2.064E+02	2.066E+02	2.067E+02	2.069E+02	2.071E+02	2.073E+02	2.076E+02	2.078E+02
PU236	0.	4.150E+02	3.271E+02	2.565E+02	2.011E+02	1.577E+02	1.236E+02	9.694E+01	7.601E+01	5.960E+01	4.673E+01	3.664E+01
PU238	0.	2.612E+06	2.754E+06	2.760E+06	2.745E+06	2.725E+06	2.704E+06	2.683E+06	2.662E+06	2.642E+06	2.621E+06	2.601E+06
PU239	0.	2.322E+05	2.368E+05	2.368E+05	2.368E+05	2.368E+05	2.368E+05	2.368E+05	2.368E+05	2.368E+05	2.367E+05	2.367E+05
PU240	0.	3.488E+05	3.490E+05	3.492E+05	3.493E+05	3.495E+05	3.496E+05	3.497E+05	3.498E+05	3.500E+05	3.501E+05	3.502E+05
PU241	0.	6.065E-08	5.783E-08	5.515E-08	5.260E-08	5.016E-08	4.783E-08	4.561E-08	4.350E-08	4.148E-08	3.956E-08	3.772E-08
PU242	0.	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02	8.576E+02
PU244	0.	1.051E-12	5.455E-12	5.861E-12	1.427E-11	1.867E-11	2.308E-11	2.748E-11	3.189E-11	3.630E-11	4.070E-11	4.511E-11
AM241	0.	7.435E+04	2.331E+05	3.842E+05	5.281E+05	6.651E+05	7.954E+05	9.194E+05	1.037E+06	1.150E+06	1.257E+06	1.358E+06
AM243	0.	1.671E+04	1.674E+04	1.674E+04	1.674E+04	1.674E+04	1.674E+04	1.674E+04	1.674E+04	1.673E+04	1.673E+04	1.673E+04
CM242	0.	4.878E+07	1.040E+07	2.209E+06	4.750E+05	1.082E+05	3.055E+04	1.409E+04	1.057E+04	9.791E+03	5.592E+03	5.516E+03
CM243	0.	5.274E+03	5.161E+03	5.050E+03	4.942E+03	4.836E+03	4.732E+03	4.631E+03	4.532E+03	4.434E+03	4.339E+03	4.246E+03
CM244	0.	3.051E+06	2.937E+06	2.827E+06	2.721E+06	2.618E+06	2.520E+06	2.425E+06	2.334E+06	2.247E+06	2.162E+06	2.081E+06
CM245	0.	2.974E+02	2.974E+02	2.974E+02	2.974E+02	2.973E+02	2.973E+02	2.973E+02	2.973E+02	2.972E+02	2.972E+02	2.972E+02
CM246	0.	7.127E+01	7.126E+01	7.125E+01	7.124E+01	7.123E+01	7.122E+01	7.121E+01	7.120E+01	7.119E+01	7.118E+01	7.117E+01
CM247	0.	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04	2.331E-04
CM248	0.	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01	1.106E-01
CF249	0.	2.109E-03	1.233E-02	1.832E-02	2.092E-02	2.148E-02	2.189E-02	2.204E-02	2.209E-02	2.208E-02	2.206E-02	2.202E-02
CF250	0.	1.434E-01	1.369E-01	1.298E-01	1.231E-01	1.167E-01	1.107E-01	1.050E-01	9.959E-02	9.444E-02	8.957E-02	8.495E-02
CF251	0.	4.539E-04	4.535E-04	4.532E-04	4.528E-04	4.525E-04	4.521E-04	4.518E-04	4.514E-04	4.511E-04	4.507E-04	4.504E-04
CF252	0.	1.069E+00	8.223E-01	6.328E-01	4.870E-01	3.747E-01	2.884E-01	2.219E-01	1.708E-01	1.314E-01	1.011E-01	7.782E-02
CF254	0.	2.535E-03	3.861E-05	5.878E-07	8.951E-09	1.361E-10	2.247E-12	3.202E-13	2.476E-13	3.860E-13	2.984E-13	2.307E-13
ES253	0.	8.308E-03	3.741E-07	2.092E-12	1.060E-17	5.304E-23	2.650E-28	1.324E-33	6.613E-39	3.303E-44	1.650E-49	8.242E-55
TOTAL	1.073E+03	5.513E+07	1.654E+07	8.790E+06	7.079E+06	6.727E+06	6.660E+06	6.652E+06	6.655E+06	6.658E+06	6.660E+06	6.660E+06

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Table A-2 Neutron Source Strength cont'd.

ACTINIDES			PWR-CYCLIC OPERATION-33,000MWD/MTHM			AVERAGE POWER(MW)		1.70230E+01				
NUCLIDE LISTING			IST SPENT FUEL DECAY UP TO 10 YRS			AVERAGE FLUX(N/CM**2-SEC)		3.64316E+13				
SPONTANEOUS FISSION NEUT RATE-NEUT/SEC			BASIS = ONE FUEL ASSEMBLY			TOTAL BURNUP(MWD)		1.50654E+04				

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
PU236	0.	9.820E+00	7.740E+00	6.369E+00	4.759E+00	3.731E+00	2.926E+00	2.294E+00	1.799E+00	1.410E+00	1.106E+00	8.671E-01
PU238	0.	1.814E+05	1.912E+05	1.916E+05	1.906E+05	1.892E+05	1.877E+05	1.863E+05	1.348E+05	1.834E+05	1.820E+05	1.806E+05
PU240	0.	9.434E+05	9.438E+05	9.442E+05	9.446E+05	9.450E+05	9.454E+05	9.457E+05	9.461E+05	9.464E+05	9.467E+05	9.470E+05
PU242	0.	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05	3.389E+05
PU244	0.	1.130E-07	5.865E-07	1.060E-06	1.534E-06	2.007E-06	2.481E-06	2.955E-06	3.428E-06	3.902E-06	4.376E-06	4.850E-06
CM242	0.	9.927E+07	2.117E+07	4.494E+06	9.666E+05	2.201E+05	6.216E+04	2.867E+04	2.151E+04	1.992E+04	1.952E+04	1.936E+04
CM244	0.	1.781E+08	1.715E+08	1.650E+08	1.588E+08	1.529E+08	1.471E+08	1.416E+08	1.363E+08	1.312E+08	1.262E+08	1.215E+08
CM246	0.	1.117E+06	1.116E+06	1.116E+06	1.116E+06	1.116E+06	1.116E+06	1.116E+06	1.115E+06	1.115E+06	1.115E+06	1.115E+06
CM248	0.	4.783E+03	4.783E+03	4.784E+03	4.784E+03	4.784E+03	4.785E+03	4.785E+03	4.785E+03	4.785E+03	4.785E+03	4.785E+03
CM250	0.	9.604E-03	9.607E-03	9.607E-03	9.606E-03	9.606E-03	9.605E-03	9.605E-03	9.605E-03	9.604E-03	9.604E-03	9.603E-03
CF252	0.	1.428E+05	1.099E+05	8.454E+04	6.505E+04	5.006E+04	3.852E+04	2.965E+04	2.281E+04	1.756E+04	1.351E+04	1.040E+04
CF254	0.	1.380E+02	2.102E+00	3.200E-02	4.873E-04	7.410E-06	1.223E-07	1.743E-08	1.348E-08	-2.101E-08	-1.625E-08	-1.256E-08
TOTAL	0.	2.801E+08	1.954E+08	1.722E+08	1.625E+08	1.557E+08	1.498E+08	1.442E+08	1.389E+08	1.338E+08	1.288E+08	1.241E+08
TOTAL	1.073E+03	3.353E+08	2.123E+08	1.810E+08	1.695E+08	1.625E+08	1.565E+08	1.509E+08	1.456E+08	1.404E+08	1.355E+08	1.308E+08

Table A-2 Neutron Source Strength cont'd.

ACTINIDES				PWR-CYCLIC OPERATION-40,000MWD/MTHM			AVERAGE POWER(MW)		2.06340E+01			
NUCLIDE LISTING				1ST SPENT FUEL DECAY UP TO 10 YRS			AVERAGE FLUX(N/CM**2-SEC)		4.59108E+13			
ALPHA N NEUTRON RATE - NEUT/SEC				BASIS = ONE FUEL ASSEMBLY			TOTAL BURNUP(MWD)		1.82611E+04			

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
BI211	0.	1.484E-03	2.878E-03	4.103E-03	5.302E-03	6.475E-03	7.623E-03	8.748E-03	9.849E-03	1.093E-02	1.199E-02	1.302E-02
BI212	0.	6.139E-02	1.516E-01	2.594E-01	3.690E-01	4.716E-01	5.631E-01	6.419E-01	7.079E-01	7.621E-01	8.055E-01	8.356E-01
BI213	0.	1.650E-08	2.393E-09	2.575E-09	2.765E-09	2.964E-09	3.172E-09	3.389E-09	3.614E-09	3.847E-09	4.090E-09	4.341E-09
PO210	0.	2.438E-07	5.636E-07	1.045E-06	1.735E-06	2.683E-06	3.940E-06	5.556E-06	7.582E-06	1.007E-05	1.306E-05	1.660E-05
PO211	0.	7.065E-06	1.370E-05	1.953E-05	2.524E-05	3.083E-05	3.629E-05	4.165E-05	4.689E-05	5.203E-05	5.706E-05	6.199E-05
PO212	0.	6.410E+00	1.583E+01	2.708E+01	3.852E+01	4.924E+01	5.880E+01	6.702E+01	7.392E+01	7.956E+01	8.410E+01	8.766E+01
PO213	0.	1.503E-03	2.180E-04	2.346E-04	2.520E-04	2.701E-04	2.890E-04	3.088E-04	3.293E-04	3.506E-04	3.727E-04	3.955E-04
PO214	0.	3.990E-05	6.017E-05	8.727E-05	1.214E-04	1.626E-04	2.110E-04	2.669E-04	3.303E-04	4.014E-04	4.804E-04	5.673E-04
PO215	0.	2.284E-03	4.430E-03	6.316E-03	8.161E-03	9.966E-03	1.173E-02	1.347E-02	1.516E-02	1.682E-02	1.845E-02	2.004E-02
PO216	0.	3.897E+00	9.623E+00	1.647E+01	2.342E+01	2.994E+01	3.575E+01	4.075E+01	4.494E+01	4.838E+01	5.114E+01	5.330E+01
PO218	0.	1.727E-05	2.604E-05	3.777E-05	5.252E-05	7.036E-05	9.134E-05	1.155E-04	1.430E-04	1.737E-04	2.079E-04	2.455E-04
AT217	0.	8.276E-04	1.200E-04	1.292E-04	1.387E-04	1.487E-04	1.591E-04	1.700E-04	1.813E-04	1.930E-04	2.052E-04	2.178E-04
RN219	0.	1.713E-03	3.323E-03	4.737E-03	6.120E-03	7.475E-03	8.800E-03	1.010E-02	1.137E-02	1.262E-02	1.384E-02	1.503E-02
RN220	0.	2.950E+00	7.284E+00	1.247E+01	1.773E+01	2.266E+01	2.706E+01	3.085E+01	3.402E+01	3.662E+01	3.871E+01	4.035E+01
RN222	0.	1.173E-05	1.769E-05	2.566E-05	3.569E-05	4.780E-05	6.206E-05	7.849E-05	9.714E-05	1.181E-04	1.413E-04	1.666E-04
FR221	0.	5.355E-04	7.767E-05	8.357E-05	8.975E-05	9.621E-05	1.029E-04	1.100E-04	1.173E-04	1.249E-04	1.327E-04	1.409E-04
RA223	0.	9.804E-04	1.902E-03	2.711E-03	3.503E-03	4.278E-03	5.037E-03	5.780E-03	6.508E-03	7.220E-03	7.919E-03	8.604E-03
RA224	0.	2.027E+00	5.005E+00	8.564E+00	1.218E+01	1.557E+01	1.859E+01	2.119E+01	2.337E+01	2.516E+01	2.659E+01	2.772E+01
RA226	0.	7.019E-06	1.058E-05	1.535E-05	2.134E-05	2.859E-05	3.712E-05	4.695E-05	5.810E-05	7.061E-05	8.449E-05	9.977E-05
AC225	0.	3.995E-04	5.794E-05	6.234E-05	6.695E-05	7.177E-05	7.680E-05	8.204E-05	8.749E-05	9.316E-05	9.903E-05	1.051E-04
AC227	0.	2.928E-12	5.179E-12	7.381E-12	9.536E-12	1.164E-11	1.371E-11	1.573E-11	1.771E-11	1.965E-11	2.155E-11	2.341E-11
TH227	0.	9.675E-04	1.812E-03	2.583E-03	3.327E-03	4.076E-03	4.798E-03	5.506E-03	6.200E-03	6.879E-03	7.544E-03	8.196E-03
TH228	0.	1.721E+00	4.263E+00	7.290E+00	1.036E+01	1.324E+01	1.581E+01	1.802E+01	1.987E+01	2.139E+01	2.261E+01	2.356E+01
TH229	0.	3.380E-05	3.644E-05	3.921E-05	4.211E-05	4.514E-05	4.830E-05	5.160E-05	5.502E-05	5.858E-05	6.228E-05	6.610E-05
TH230	0.	6.824E-03	9.558E-03	1.235E-02	1.520E-02	1.811E-02	2.108E-02	2.411E-02	2.719E-02	3.033E-02	3.353E-02	3.678E-02
TH232	0.	7.858E-09	1.216E-08	1.646E-08	2.076E-08	2.507E-08	2.937E-08	3.367E-08	3.797E-08	4.228E-08	4.658E-08	5.089E-08
PA231	0.	1.681E-02	1.704E-02	1.720E-02	1.736E-02	1.752E-02	1.768E-02	1.784E-02	1.800E-02	1.816E-02	1.833E-02	1.849E-02
U232	0.	6.904E+00	1.132E+01	1.471E+01	1.731E+01	1.930E+01	2.080E+01	2.192E+01	2.275E+01	2.334E+01	2.376E+01	2.403E+01
U233	0.	2.354E-02	2.474E-02	2.595E-02	2.717E-02	2.838E-02	2.959E-02	3.081E-02	3.202E-02	3.324E-02	3.446E-02	3.568E-02
U234	9.307E+02	3.340E+02	3.412E+02	3.485E+02	3.557E+02	3.630E+02	3.702E+02	3.773E+02	3.844E+02	3.914E+02	3.983E+02	4.052E+02
U235	3.341E+01	5.321E+00	5.321E+00	5.321E+00	5.321E+00	5.321E+00	5.321E+00	5.321E+00	5.322E+00	5.322E+00	5.322E+00	5.322E+00
U236	0.	1.327E+02	1.327E+02	1.327E+02	1.327E+02	1.327E+02	1.327E+02	1.327E+02	1.327E+02	1.327E+02	1.328E+02	1.328E+02
U238	1.086E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02	1.051E+02
NP237	0.	2.562E+02	2.644E+02	2.644E+02	2.646E+02	2.647E+02	2.649E+02	2.651E+02	2.653E+02	2.656E+02	2.659E+02	2.662E+02
PU236	0.	7.010E+02	5.526E+02	4.333E+02	3.397E+02	2.664E+02	2.089E+02	1.638E+02	1.284E+02	1.007E+02	7.855E+01	6.191E+01
PU238	0.	4.120E+06	4.334E+06	4.342E+06	4.317E+06	4.285E+06	4.253E+06	4.220E+06	4.187E+06	4.155E+06	4.122E+06	4.090E+06
PU239	0.	2.331E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05	2.392E+05
PU240	0.	3.833E+05	3.838E+05	3.842E+05	3.847E+05	3.851E+05	3.855E+05	3.859E+05	3.863E+05	3.867E+05	3.870E+05	3.874E+05
PU241	0.	7.323E-08	6.983E-08	6.660E-08	6.351E-08	6.056E-08	5.776E-08	5.508E-08	5.252E-08	5.009E-08	4.777E-08	4.555E-08
PU242	0.	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03	1.335E+03
PU244	0.	7.477E-12	3.887E-11	7.026E-11	1.017E-10	1.331E-10	1.645E-10	1.959E-10	2.273E-10	2.587E-10	2.901E-10	3.215E-10
AM241	0.	7.684E+04	2.686E+05	4.511E+05	6.248E+05	7.902E+05	9.476E+05	1.097E+06	1.240E+06	1.376E+06	1.505E+06	1.627E+06
AM243	0.	3.301E+04	3.307E+04	3.306E+04	3.306E+04	3.306E+04	3.306E+04	3.305E+04	3.305E+04	3.305E+04	3.304E+04	3.304E+04
CM242	0.	7.219E+07	1.538E+07	3.263E+06	6.987E+05	1.562E+05	4.136E+04	1.703E+04	1.184E+04	1.071E+04	1.043E+04	1.033E+04
CM243	0.	9.321E+03	9.121E+03	8.925E+03	8.734E+03	8.547E+03	8.364E+03	8.185E+03	8.009E+03	7.838E+03	7.670E+03	7.505E+03
CM244	0.	8.065E+06	7.763E+06	7.471E+06	7.191E+06	6.921E+06	6.661E+06	6.410E+06	6.169E+06	5.938E+06	5.715E+06	5.500E+06
CM245	0.	9.212E+02	9.211E+02	9.211E+02	9.210E+02	9.209E+02	9.208E+02	9.208E+02	9.207E+02	9.206E+02	9.205E+02	9.204E+02
CM246	0.	3.005E+02	3.005E+02	3.004E+02	3.004E+02	3.004E+02	3.003E+02	3.003E+02	3.002E+02	3.002E+02	3.001E+02	3.001E+02
CM247	0.	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03	1.246E-03
CM248	0.	7.880E-01	7.882E-01	7.883E-01	7.884E-01	7.885E-01	7.886E-01	7.886E-01	7.887E-01	7.887E-01	7.887E-01	7.887E-01
BK249	0.	1.032E-11	4.615E-12	2.061E-12	9.202E-13	4.109E-13	1.835E-13	8.192E-14	3.658E-14	1.633E-14	7.293E-15	3.256E-15
CF249	0.	1.623E-02	1.148E-01	1.587E-01	1.780E-01	1.864E-01	1.900E-01	1.914E-01	1.918E-01	1.917E-01	1.915E-01	1.912E-01
CF250	0.	1.401E+00	1.338E+00	1.269E+00	1.204E+00	1.142E+00	1.083E+00	1.027E+00	9.738E-01	9.235E-01	8.758E-01	8.306E-01
CF251	0.	4.882E-03	4.878E-03	4.874E-03	4.870E-03	4.867E-03	4.863E-03	4.859E-03	4.855E-03	4.852E-03	4.848E-03	4.844E-03
CF252	0.	1.531E+01	1.178E+01	9.064E+00	6.975E+00	5.368E+00	4.131E+00	3.179E+00	2.446E+00	1.882E+00	1.449E+00	1.115E+00
CF254	0.	6.221E-02	9.472E-04	1.442E-05	2.196E-07	3.344E-09	5.120E-11	5.484E-13	1.707E-13	2.662E-13	2.056E-13	3.209E-13
ES253	0.	1.554E-01	7.030E-06	3.931E-11	1.992E-16	9.967E-22	4.980E-27	2.488E-32	1.243E-37	6.207E-43	3.101E-48	1.549E-53
TOTAL	1.073E+03	8.512E+07	2.842E+07	1.620E+07	1.350E+07	1.282E+07	1.257E+07	1.241E+07	1.228E+07	1.215E+07	1.202E+07	1.190E+07

Table A-2 Neutron Source Strength cont'd.

ACTINIDES		FWR-CYCLIC OPERATION-40,000MWD/MTHM	AVERAGE POWER(MW)	2.06340E+01
NUCLIDE LISTING		1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	4.59108E+13
SPONTANEOUS FISSION NEUT RATE-NEUT/SEC		BASIS = ONE FUEL ASSEMBLY	TOTAL BURNUP(MWD)	1.82611E+04

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
PU236	0.	1.659E+01	1.308E+01	1.025E+01	8.039E+00	6.303E+00	4.943E+00	3.875E+00	3.039E+00	2.383E+00	1.868E+00	1.465E+00
PU238	0.	2.860E+05	3.010E+05	3.015E+05	2.997E+05	2.975E+05	2.953E+05	2.930E+05	2.907E+05	2.885E+05	2.862E+05	2.840E+05
PU240	0.	1.036E+06	1.038E+06	1.039E+06	1.040E+06	1.042E+06	1.043E+06	1.044E+06	1.045E+06	1.046E+06	1.047E+06	1.048E+06
PU242	0.	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05	5.275E+05
PU244	0.	8.039E-07	4.178E-06	7.554E-06	1.093E-05	1.431E-05	1.768E-05	2.106E-05	2.444E-05	2.781E-05	3.119E-05	3.457E-05
CM242	0.	1.469E+08	3.130E+07	6.640E+06	1.422E+06	3.178E+05	8.416E+04	3.465E+04	2.409E+04	2.179E+04	2.122E+04	2.103E+04
CM244	0.	4.708E+08	4.532E+08	4.362E+08	4.198E+08	4.040E+08	3.889E+08	3.742E+08	3.602E+08	3.467E+08	3.336E+08	3.211E+08
CM246	0.	4.708E+06	4.708E+06	4.707E+06	4.706E+06	4.706E+06	4.705E+06	4.704E+06	4.704E+06	4.703E+06	4.702E+06	4.701E+06
CM248	0.	3.408E+04	3.409E+04	3.410E+04	3.410E+04	3.410E+04	3.411E+04	3.411E+04	3.411E+04	3.411E+04	3.411E+04	3.411E+04
CM250	0.	1.210E-01	1.211E-01	1.211E-01	1.211E-01	1.211E-01	1.210E-01	1.210E-01	1.210E-01	1.210E-01	1.210E-01	1.210E-01
CF252	0.	2.045E+06	1.574E+06	1.211E+06	9.319E+05	7.171E+05	5.518E+05	4.247E+05	3.268E+05	2.515E+05	1.935E+05	1.489E+05
CF254	0.	3.387E+03	5.157E+01	7.852E-01	1.196E-02	1.820E-04	2.787E-06	2.986E-03	-9.294E-09	1.449E-08	1.120E-08	-1.747E-08
TOTAL	0.	6.264E+08	4.527E+08	4.507E+08	4.288E+08	4.117E+08	3.961E+08	3.813E+08	3.671E+08	3.535E+08	3.404E+08	3.275E+08
TOTAL	1.073E+03	7.115E+08	5.213E+08	4.669E+08	4.423E+08	4.245E+08	4.087E+08	3.937E+08	3.794E+08	3.657E+08	3.525E+08	3.358E+08

ACTINIDES	GE STD BWR 6-4 CYCLE=2.66 W/C HIGH	AVERAGE POWER(MW)	5.65050E+00
NUCLIDE LISTING	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.42695E+13
ALPHA N NEUTRON RATE - NEUT/SEC	BASIS= ONE FUEL ASSEMBLY	TOTAL BURNUP(MWD)	6.21555E+03

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
BI211	0.	1.153E-03	1.943E-03	2.634E-03	3.308E-03	3.965E-03	4.607E-03	5.234E-03	5.845E-03	6.443E-03	7.026E-03	7.596E-03
BI212	0.	5.267E-02	1.011E-01	1.558E-01	2.098E-01	2.596E-01	3.035E-01	3.410E-01	3.722E-01	3.976E-01	4.178E-01	4.336E-01
BI213	0.	8.759E-09	1.545E-09	1.631E-09	1.720E-09	1.814E-09	1.913E-09	2.016E-09	2.123E-09	2.235E-09	2.352E-09	2.473E-09
PO210	0.	2.997E-07	5.724E-07	9.528E-07	1.465E-06	2.126E-06	2.990E-06	4.054E-06	5.354E-06	6.913E-06	8.758E-06	1.091E-05
PO211	0.	5.491E-06	5.251E-06	1.254E-05	1.575E-05	1.888E-05	2.193E-05	2.492E-05	2.783E-05	3.067E-05	3.345E-05	3.617E-05
PO212	0.	5.499E+00	1.056E+01	1.626E+01	2.191E+01	2.711E+01	3.169E+01	3.560E+01	3.886E+01	4.151E+01	4.363E+01	4.528E+01
PO213	0.	7.981E-04	1.408E-04	1.486E-04	1.567E-04	1.653E-04	1.743E-04	1.837E-04	1.935E-04	2.037E-04	2.143E-04	2.253E-04
PO214	0.	3.425E-05	4.815E-05	6.566E-05	8.683E-05	1.117E-04	1.405E-04	1.731E-04	2.097E-04	2.504E-04	2.951E-04	3.440E-04
PO215	0.	1.775E-03	2.991E-03	4.054E-03	5.091E-03	6.104E-03	7.092E-03	8.056E-03	8.998E-03	9.917E-03	1.082E-02	1.169E-02
PO216	0.	3.349E+00	6.419E+00	9.889E+00	1.332E+01	1.648E+01	1.927E+01	2.165E+01	2.363E+01	2.529E+01	2.653E+01	2.753E+01
PO218	0.	1.482E-05	2.084E-05	2.842E-05	3.758E-05	4.837E-05	6.081E-05	7.493E-05	9.077E-05	1.084E-04	1.277E-04	1.489E-04
AT217	0.	4.394E-04	7.751E-05	8.179E-05	8.629E-05	9.101E-05	5.596E-05	1.011E-04	1.065E-04	1.121E-04	1.180E-04	1.240E-04
RN219	0.	1.332E-03	2.243E-03	3.040E-03	3.818E-03	4.578E-03	5.319E-03	6.042E-03	6.748E-03	7.438E-03	8.111E-03	8.769E-03
RN220	0.	2.531E+00	4.859E+00	7.486E+00	1.008E+01	1.248E+01	1.459E+01	1.639E+01	1.789E+01	1.911E+01	2.008E+01	2.084E+01
RN222	0.	1.007E-05	1.416E-05	1.931E-05	2.553E-05	3.286E-05	4.131E-05	5.091E-05	6.167E-05	7.362E-05	8.678E-05	1.012E-04
FR221	0.	2.843E-04	5.015E-05	5.292E-05	5.583E-05	5.889E-05	6.208E-05	6.543E-05	6.892E-05	7.255E-05	7.633E-05	8.025E-05
RA223	0.	7.621E-04	1.284E-03	1.740E-03	2.185E-03	2.620E-03	3.044E-03	3.458E-03	3.862E-03	4.257E-03	4.642E-03	5.019E-03
RA224	0.	1.739E+00	3.338E+00	5.143E+00	6.927E+00	8.572E+00	1.002E+01	1.126E+01	1.229E+01	1.313E+01	1.380E+01	1.432E+01
RA226	0.	6.024E-06	8.470E-06	1.155E-05	1.527E-05	1.966E-05	2.471E-05	3.045E-05	3.689E-05	4.403E-05	5.190E-05	6.051E-05
AC225	0.	2.121E-04	3.742E-05	3.948E-05	4.165E-05	4.393E-05	4.632E-05	4.881E-05	5.141E-05	5.412E-05	5.694E-05	5.987E-05
AC227	0.	2.223E-12	3.496E-12	4.738E-12	5.949E-12	7.131E-12	8.285E-12	9.412E-12	1.051E-11	1.159E-11	1.263E-11	1.366E-11
TH227	0.	7.450E-04	1.223E-03	1.658E-03	2.082E-03	2.496E-03	2.900E-03	3.294E-03	3.679E-03	4.055E-03	4.423E-03	4.782E-03
TH228	0.	1.476E+00	2.842E+00	4.377E+00	5.893E+00	7.290E+00	8.522E+00	9.573E+00	1.045E+01	1.116E+01	1.173E+01	1.217E+01
TH229	0.	2.230E-05	2.353E-05	2.483E-05	2.619E-05	2.763E-05	2.913E-05	3.070E-05	3.233E-05	3.404E-05	3.581E-05	3.765E-05
TH230	0.	4.900E-03	6.340E-03	7.810E-03	9.310E-03	1.084E-02	1.240E-02	1.399E-02	1.561E-02	1.726E-02	1.894E-02	2.065E-02
TH232	0.	6.778E-09	8.939E-09	1.110E-08	1.326E-08	1.542E-08	1.759E-08	1.975E-08	2.191E-08	2.407E-08	2.623E-08	2.840E-08
PA231	0.	9.661E-03	9.752E-03	9.817E-03	9.882E-03	9.947E-03	1.001E-02	1.008E-02	1.014E-02	1.021E-02	1.027E-02	1.034E-02
U232	0.	4.404E+00	6.453E+00	8.025E+00	9.229E+00	1.014E+01	1.084E+01	1.135E+01	1.172E+01	1.199E+01	1.217E+01	1.229E+01
U233	0.	1.096E-02	1.158E-02	1.220E-02	1.282E-02	1.345E-02	1.407E-02	1.469E-02	1.531E-02	1.594E-02	1.656E-02	1.718E-02
U234	4.300E+02	1.759E+02	1.796E+02	1.834E+02	1.872E+02	1.909E+02	1.946E+02	1.983E+02	2.020E+02	2.056E+02	2.092E+02	2.128E+02
U235	1.080E+01	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00	2.148E+00
U236	3.164E+01	6.667E+01	6.668E+01	6.668E+01	6.668E+01	6.668E+01	6.669E+01	6.669E+01	6.669E+01	6.669E+01	6.670E+01	6.670E+01
U238	4.368E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01	4.243E+01
NP237	0.	1.331E+02	1.355E+02	1.355E+02	1.356E+02	1.356E+02	1.357E+02	1.358E+02	1.359E+02	1.360E+02	1.361E+02	1.362E+02
PU236	0.	3.280E+02	2.581E+02	2.023E+02	1.587E+02	1.244E+02	9.755E+01	7.649E+01	5.997E+01	4.702E+01	3.687E+01	2.891E+01
PU238	0.	2.178E+06	2.250E+06	2.248E+06	2.234E+06	2.217E+06	2.200E+06	2.183E+06	2.166E+06	2.149E+06	2.133E+06	2.116E+06
PU239	0.	9.457E+04	9.601E+04	9.600E+04	9.600E+04	9.600E+04	9.600E+04	9.599E+04	9.599E+04	9.599E+04	9.599E+04	9.598E+04
PU240	0.	1.481E+05	1.482E+05	1.483E+05	1.484E+05	1.485E+05	1.486E+05	1.487E+05	1.488E+05	1.488E+05	1.489E+05	1.490E+05
PU241	0.	2.675E-08	2.551E-08	2.433E-08	2.320E-08	2.212E-08	2.110E-08	2.012E-08	1.919E-08	1.830E-08	1.745E-08	1.664E-08
PU242	0.	4.309E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02	4.310E+02
PU244	0.	1.745E-12	6.355E-12	1.097E-11	1.558E-11	2.019E-11	2.480E-11	2.941E-11	3.402E-11	3.863E-11	4.325E-11	4.786E-11
AM241	0.	4.761E+04	1.176E+05	1.842E+05	2.477E+05	3.081E+05	3.655E+05	4.202E+05	4.722E+05	5.218E+05	5.688E+05	6.136E+05
AM243	0.	9.468E+03	9.480E+03	9.479E+03	9.478E+03	9.477E+03	9.476E+03	9.475E+03	9.474E+03	9.474E+03	9.473E+03	9.472E+03
CM242	0.	2.722E+07	5.801E+06	1.232E+06	2.658E+05	6.125E+04	1.796E+04	8.779E+03	6.814E+03	6.375E+03	6.260E+03	6.213E+03
CM243	0.	3.150E+03	3.082E+03	3.016E+03	2.952E+03	2.888E+03	2.826E+03	2.766E+03	2.707E+03	2.649E+03	2.592E+03	2.536E+03
CM244	0.	1.971E+06	1.897E+06	1.826E+06	1.758E+06	1.691E+06	1.628E+06	1.567E+06	1.508E+06	1.451E+06	1.397E+06	1.344E+06
CM245	0.	2.080E+02	2.080E+02	2.080E+02	2.079E+02	2.079E+02	2.079E+02	2.079E+02	2.079E+02	2.079E+02	2.078E+02	2.078E+02
CM246	0.	5.783E+01	5.782E+01	5.781E+01	5.780E+01	5.779E+01	5.778E+01	5.778E+01	5.777E+01	5.776E+01	5.775E+01	5.774E+01
CM247	0.	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04	2.124E-04
CM248	0.	1.157E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01
CF249	0.	3.405E-03	1.565E-02	2.109E-02	2.349E-02	2.453E-02	2.497E-02	2.514E-02	2.519E-02	2.518E-02	2.515E-02	2.511E-02
CF250	0.	1.661E-01	1.583E-01	1.501E-01	1.424E-01	1.350E-01	1.280E-01	1.214E-01	1.152E-01	1.092E-01	1.036E-01	9.823E-02
CF251	0.	5.518E-04	5.513E-04	5.509E-04	5.505E-04	5.501E-04	5.497E-04	5.492E-04	5.488E-04	5.484E-04	5.480E-04	5.475E-04
CF252	0.	1.459E+00	1.122E+00	8.637E-01	6.647E-01	5.115E-01	3.936E-01	3.029E-01	2.331E-01	1.794E-01	1.380E-01	1.062E-01
CF254	0.	2.751E-03	4.188E-05	6.378E-07	9.711E-09	1.477E-10	2.437E-12	3.474E-13	2.686E-13	2.077E-13	-3.238E-13	-2.503E-13
ES253	0.	1.078E-02	4.450E-07	2.485E-12	1.259E-17	6.301E-23	3.148E-28	1.573E-33	7.855E-39	3.924E-44	1.960E-49	9.791E-55
TOTAL	5.161E+02	3.168E+07	1.032E+07	5.745E+06	4.763E+06	4.536E+06	4.470E+06	4.437E+06	4.411E+06	4.387E+06	4.363E+06	4.335E+06

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Table A-2 Neutron Source Strength cont'd.

ACTINIDES		GE STD BWR 6-4 CYCLE-2.66 W/C HIGH		AVERAGE POWER(MW)		5.65050E+00
NUCLIDE LISTING		1ST SPENT FUEL DECAY UP TO 10 YRS		AVERAGE FLUX(N/CM**2-SEC)		3.42695E+13
SPONTANEOUS FISSION NEUT RATE-NEUT/SEC		BASIS= ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)		6.21555E+03

NUCLIDE	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
PU236	0.	7.761E+00	6.106E+00	4.788E+00	3.754E+00	2.944E+00	2.308E+00	1.810E+00	1.419E+00	1.113E+00	8.725E-01	6.841E-01
PU238	0.	1.512E+05	1.562E+05	1.561E+05	1.551E+05	1.539E+05	1.528E+05	1.516E+05	1.504E+05	1.492E+05	1.481E+05	1.469E+05
PU240	0.	4.004E+05	4.007E+05	4.010E+05	4.013E+05	4.016E+05	4.018E+05	4.021E+05	4.023E+05	4.025E+05	4.028E+05	4.030E+05
PU242	0.	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05	1.703E+05
PU244	0.	1.876E-07	6.832E-07	1.179E-06	1.675E-06	2.170E-06	2.666E-06	3.162E-06	3.658E-06	4.153E-06	4.649E-06	5.145E-06
CM242	0.	5.540E+07	1.180E+07	2.508E+06	5.408E+05	1.246E+05	3.655E+04	1.786E+04	1.387E+04	1.297E+04	1.274E+04	1.264E+04
CM244	0.	1.151E+08	1.108E+08	1.066E+08	1.026E+03	9.875E+07	9.504E+07	9.147E+07	8.803E+07	8.473E+07	8.154E+07	7.848E+07
CM246	0.	9.060E+05	9.058E+05	9.057E+05	9.056E+05	9.054E+05	9.053E+05	9.052E+05	9.050E+05	9.049E+05	9.048E+05	9.046E+05
CM248	0.	5.006E+03	5.006E+03	5.007E+03	5.007E+03	5.008E+03	5.008E+03	5.008E+03	5.008E+03	5.008E+03	5.009E+03	5.009E+03
CM250	0.	9.031E-03	9.034E-03	9.034E-03	9.034E-03	9.033E-03	9.033E-03	9.033E-03	9.032E-03	9.032E-03	9.032E-03	9.031E-03
CF252	0.	1.949E+05	1.500E+05	1.154E+05	8.880E+04	6.833E+04	5.259E+04	4.047E+04	3.114E+04	2.396E+04	1.844E+04	1.419E+04
CF254	0.	1.498E+02	2.280E+00	3.472E-02	5.287E-04	8.040E-06	1.327E-07	1.891E-08	1.462E-08	1.131E-08	-1.763E-08	-1.363E-08
TOTAL	0.	1.723E+08	1.244E+08	1.189E+08	1.049E+08	1.006E+08	9.677E+07	9.316E+07	8.971E+07	8.640E+07	8.320E+07	8.014E+07
TOTAL	5.161E+02	2.040E+08	1.347E+08	1.156E+08	1.096E+08	1.051E+08	1.012E+08	9.760E+07	9.412E+07	9.078E+07	8.757E+07	8.447E+07

Table A-3 Structural Material Gamma Source Strengths

STRUCTURAL MATERIALS	PWR-CYCLIC OPERATION-20,000MWD/MTHM	AVERAGE POWER(MW)	1.54956E+01
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.08511E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV BASIS = ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	9.14240E+03

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	9.450E+12	6.231E+09	4.105E+09	3.139E+09	2.478E+09	1.928E+09	1.501E+09	1.170E+09	9.138E+08	7.153E+08	5.616E+08
.63	0.	9.443E+14	2.339E+13	2.659E+12	1.035E+12	4.83cE+11	2.401E+11	1.281E+11	7.463E+10	4.757E+10	3.285E+10	2.415E+10
1.10	0.	1.053E+13	1.338E+12	1.151E+12	1.009E+12	8.842E+11	7.750E+11	6.793E+11	5.955E+11	5.219E+11	4.575E+11	4.010E+11
1.55	0.	6.780E+12	2.593E+10	7.379E+08	2.107E+07	6.068E+05	2.111E+04	4.235E+03	3.662E+03	3.560E+03	3.472E+03	3.388E+03
1.99	0.	3.315E+14	7.787E+07	1.161E+06	1.747E+04	4.113E+02	1.532E+02	1.458E+02	1.422E+02	1.387E+02	1.353E+02	1.320E+02
2.38	0.	1.083E+13	1.464E-04	5.913E-16	2.392E-27	9.668E-39	3.908E-50	1.579E-61	6.384E-73	2.580E-84	1.043E-95	4.216-107
2.75	0.	1.157E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.25	0.	2.677E+12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.70	0.	9.574E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	2.024E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	5.442E+12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	1.333E+15	2.476E+13	3.315E+12	2.047E+12	1.370E+12	1.017E+12	8.090E+11	6.713E+11	5.704E+11	4.911E+11	4.257E+11

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STRUCTURAL MATERIALS	PWR-CYCLIC OPERATION-33,000MWD/MTHM	AVERAGE POWER(MW)	1.70230E+01
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.64316E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV BASIS = ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	1.50654E+04

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	2.737E+13	1.176E+10	7.111E+09	5.523E+09	4.293E+09	3.340E+09	2.601E+09	2.028E+09	1.585E+09	1.241E+09	9.750E+08
.63	0.	2.160E+15	4.527E+13	4.372E+12	1.641E+12	7.713E+11	3.871E+11	2.096E+11	1.240E+11	8.022E+10	5.607E+10	4.156E+10
1.10	0.	2.824E+13	3.746E+12	3.229E+12	2.830E+12	2.481E+12	2.175E+12	1.906E+12	1.671E+12	1.465E+12	1.284E+12	1.125E+12
1.55	0.	2.411E+13	4.729E+10	1.337E+09	3.808E+07	1.100E+06	4.251E+04	1.196E+04	1.083E+04	1.054E+04	1.028E+04	1.003E+04
1.59	0.	1.046E+15	2.077E+08	3.097E+06	4.665E+04	1.142E+03	4.525E+02	4.316E+02	4.209E+02	4.107E+02	4.006E+02	3.905E+02
2.38	0.	3.418E+13	4.679E-04	1.891E-15	7.644E-27	3.090E-38	1.249E-49	5.048E-61	2.040E-72	8.247E-84	3.333E-95	1.347-106
2.75	0.	3.653E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.25	0.	8.450E+12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.70	0.	5.947E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	1.069E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	1.742E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	3.382E+15	4.907E+13	7.610E+12	4.477E+12	3.256E+12	2.565E+12	2.118E+12	1.797E+12	1.546E+12	1.341E+12	1.168E+12

Table A-3 Structural Material Gamma Source Strengths cont'd.

STRUCTURAL MATERIALS	PWR-CYCLIC OPERATION-4C,C00MWD/MTHM	AVERAGE POWER(MW)	2.06340E+C1
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	4.59108E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV BASIS = ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	1.82611E+04

EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	3.632E+13	1.546E+10	9.326E+09	7.243E+09	5.630E-09	4.380E+09	3.410E+09	2.659E+09	2.077E+09	1.626E+09	1.277E+09
.63	0.	2.841E+15	5.933E+13	5.693E+12	2.136E+12	1.039E+12	5.043E+11	2.733E+11	1.618E+11	1.047E+11	7.319E+10	5.425E+10
1.10	0.	3.973E+13	6.407E+12	5.543E+12	4.859E+12	4.259E+12	3.733E+12	3.272E+12	2.868E+12	2.514E+12	2.204E+12	1.932E+12
1.55	0.	3.500E+13	5.667E+10	1.559E+09	4.550E+07	1.313E+06	5.567E+04	1.909E+04	1.761E+04	1.715E+04	1.673E+04	1.633E+04
1.99	0.	1.385E+15	2.752E+08	4.164E+06	6.284E+04	1.664E+03	7.336E+02	7.025E+02	6.852E+02	6.685E+02	6.522E+02	6.363E+02
2.38	0.	4.524E+13	6.248E-04	2.525E-15	1.021E-26	4.125E-38	1.668E-49	6.741E-61	2.725E-72	1.101E-83	4.451E-95	1.759-106
2.75	0.	4.835E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.25	0.	1.118E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.70	0.	2.156E-01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	1.774E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	2.327E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	4.464E+15	6.581E+13	1.125E+13	7.002E+12	5.269E+12	4.242E+12	3.549E+12	3.033E+12	2.621E+12	2.278E+12	1.987E+12

STRUCTURAL MATERIALS	GE STD BWR 6-4 CYCLE-2.66 W/C HIGH	AVERAGE POWER(MW)	5.65050E+00
GAMMA PHOTON SOURCE SPECTRUM	1ST SPENT FUEL DECAY UP TO 10 YRS	AVERAGE FLUX(N/CM**2-SEC)	3.42695E+13
TOTAL ENERGY RATE - MEV/SEC EMEAN=MEV BASIS= ONE FUEL ASSEMBLY		TOTAL BURNUP(MWD)	6.21555E+03

| EMEAN	INPUT	INITIAL	1.0 Y	2.0 Y	3.0 Y	4.0 Y	5.0 Y	6.0 Y	7.0 Y	8.0 Y	9.0 Y	10.0 Y
.30	0.	8.405E+12	4.956E+09	3.026E+09	2.351E+09	1.828E+09	1.422E+09	1.108E+09	8.645E+08	6.759E+08	5.298E+08	4.167E+08
.63	0.	9.857E+14	1.591E+13	2.253E+12	9.101E+11	4.155E+11	2.029E+11	1.043E+11	5.786E+10	3.490E+10	2.274E+10	1.576E+10
1.10	0.	1.068E+13	1.078E+12	9.170E+11	8.037E+11	7.044E+11	6.175E+11	5.412E+11	4.744E+11	4.158E+11	3.645E+11	3.195E+11
1.55	0.	7.296E+12	7.745E+09	2.133E+08	5.993E+06	1.758E+05	1.066E+04	5.813E+03	5.540E+03	5.401E+03	5.270E+03	5.141E+03
1.99	0.	6.496E+14	5.189E+07	1.371E+06	2.067E+04	5.371E+02	2.312E+02	2.212E+02	2.158E+02	2.105E+02	2.054E+02	2.004E+02
2.38	0.	2.128E+13	1.558E-04	6.299E-16	2.546E-27	1.029E-38	4.159E-50	1.681E-61	6.795E-73	2.747E-84	1.110E-95	4.487-107
2.75	0.	2.268E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.25	0.	5.259E+12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.70	0.	4.998E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22	0.	3.700E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25	0.	6.297E-21	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	1.711E+15	1.700E+13	3.173E+12	1.716E+12	1.126E+12	8.218E+11	6.466E+11	5.331E+11	4.514E+11	3.878E+11	3.357E+11

APPENDIX B

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 15 1979
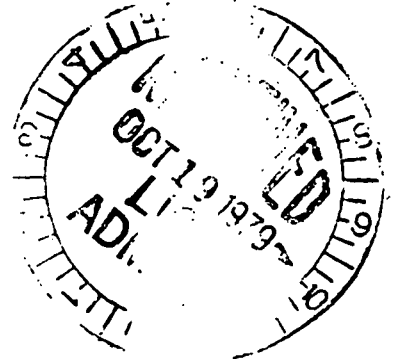
General Atomic Company
ATTN: Mr. William R. Mowry
P. O. Box 81608
San Diego, CA 92138

Gentlemen:

Enclosed is Certificate of Compliance No. 6346, Revision No. 3, for the Model No. FSV-1 shipping container. This certificate supersedes, in its entirety, Certificate of Compliance No. 6346, Revision 2, dated August 25, 1977.

Changes made to the enclosed certificate are indicated by vertical lines in the margin.

General Atomic Company and Public Service Company of Colorado have been registered as users of this package under the general license provisions of 10 CFR §71.12(b) or 49 CFR §173.393a.

This approval constitutes authority to use this package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR §173.393a.

Sincerely,

Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and
Material Safety

Enclosure: As stated

cc: w/encl
Mr. Richard R. Rawl
Department of Transportation

Public Service Company of Colorado
ATTN: Mr. J. K. Fuller
P.O. Box 361
Platteville, CO 80651

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE

For Radioactive Materials Packages

1.(a) Certificate Number	1.(b) Revision No.	1.(c) Package Identification No.	1.(d) Pages No.	1.(e) Total No. Pages
6346	3	USA/6346/B( )F	1	3

2. PREAMBLE

2.(a) This certificate is issued to satisfy Sections 173.393a, 173.394, 173.395, and 173.396 of the Department of Transportation Hazardous Materials Regulations (49 CFR 170-189 and 14 CFR 103) and Sections 146—19—10a and 146—19—100 of the Department of Transportation Dangerous Cargoes Regulations (46 CFR 146—149), as amended.

2.(b) The packaging and contents described in item 5 below, meets the safety standards set forth in Subpart C of Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Material Under Certain Conditions."

2.(c) This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. This certificate is issued on the basis of a safety analysis report of the package design or application—

3.(a) Prepared by (Name and address):

General Atomic Company
P. O. Box 81608
San Diego, CA 92136

3.(b) Title and identification of report or application:

Gulf General Atomic, Inc. application dated May 20, 1969, as supplemented.

3.(c) Docket No. 71-6346

4. CONDITIONS

This certificate is conditional upon the fulfilling of the requirements of Subpart D of 10 CFR 71, as applicable, and the conditions specified in item 5 below.

5. Description of Packaging and Authorized Contents, Model Number, Fissile Class, Other Conditions, and References:

(a) Packaging

   (1) Model No.: FSV-1

   (2) Description

The cask is cylindrical, 208" long and 28" diameter for most of its length except for the top flange which is 11-3/8" thick and 31" in diameter. The cask utilizes stainless steel encased depleted uranium as primary shielding. The cask's inner container, for fueled contents, has a cavity of 16-5/8" diameter and 187/-5/8" long. The fuel contents are hexagonal cross section elements and are loaded into the cask's inner container, six in one column. The cask may utilize either of two closure configurations. One closure system utilizes an outer cover consisting of 3-3/4" stainless steel together depleted uranium, 2-1/4" thick, and an inner lid of 2-1/4" thick stainless steel. The primary seal is provided by two O-rings, one metallic and the other silicone. The alternate closure system utilizes 4.75" thick alloy steel and 4.14" of depleted uranium. The alternate closure system requires the use of an impact limiter at the top during transport. The alternate closure is sealed by a molded dual concentric silicone elastomer seal ring with a test port between the seals. A gas sample port penetrates through the lid into the container. The gas test port is sealed with a check valve and primary plug with a silicone O-ring.

(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with National Lead Company Drawings Nos.: 40065D, Rev. 1; 70085F, Rev. 5; 70086F, Rev. 7; 70094F, Rev. 5; 40155D, Rev. 1; 70105F, Rev. 5; 70104F, Rev. 5; and 70296F, Rev. 2.

The packaging with its alternate closure system utilizes the above drawings except that closures with impact limiter are constructed as shown on General Atomic Company Drawings Nos. GADR 55-2-1, Issue A; GADR 55-2-2, Issue A; and GADR 55-2-3, Issue A.

(b) Contents

(1) Type and form of material

(i) Irradiated fuel elements consisting of graphite body, hexagonal in horizontal cross section, approximately 31.2 inches high and 14.2 inches across the flats. Each fuel element prior to irradiation contains thorium and uranium enriched to a maximum of 93.5 w/o in the U-235 isotope.

(ii) Solid nonfissile irradiated hardware and neutron source components.

(2) Maximum quantity of material per package

Not to exceed a decay heat generation of 4.1 kw and:

(i) Item 5(b)(1)(i) above:

Six fuel elements each containing 1.4 kg of enriched uranium and 11.3 kg of thorium prior to irradiation and weighing 300 lbs;

(ii) Item 5(b)(1)(ii) above:

As needed, appropriate component spacers shall be used in the cask cavity to limit movement of contents during shipment.

(c) Fissile Class   III

Maximum number of packages per shipment   One (1)

6. The alternate closure system shall be leak tested prior to (1) first use and periodically in accordance with Section I.G.2 and (2) each use in accordance with Section I.G.1, to Addendum I, General Atomic Company Safety Analysis Report GADR-55.

(

7. The spent fuel container main flange seal(s) and the alternate closure system fuel container primary plug silicone O-ring shall be replaced at least every 12 months.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12(b).

9. Expiration date: October 31, 1980.

REFERENCES

Gulf General Atomic, Inc. application dated May 20, 1969.

Supplements dated: December 2, 1969; June 17 and July 27, 1970; August 3, 1971; June 15, 1977; August 15, 1978; and February 27, April 6, September 6, and October 12, 1979.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and
  Material Safety

Date: OCT 15 1979

APPENDIX C

NOTES:
1. CASK AND CONTAINER BODY CONFIGURA-
   TIONS PER NL DWG 70086F Rev. 7,
   70094F Rev. 5, 70296F Rev. 2, 70085F Rev. 5
   AND 400G5D Rev. 1.

1¼-7 BOLT × 4½ LONG
(24 PLACES)
AMS 5737 STEEL

⅝-11 THREAD
× .62 LONG

RETAINER RING
6061 AL ALLOY
33.12 O.D. × 28.81 I.O. × 1.0 THICK

13.50

¼ 6061 AL ALLOY

SILICONE RUBBER
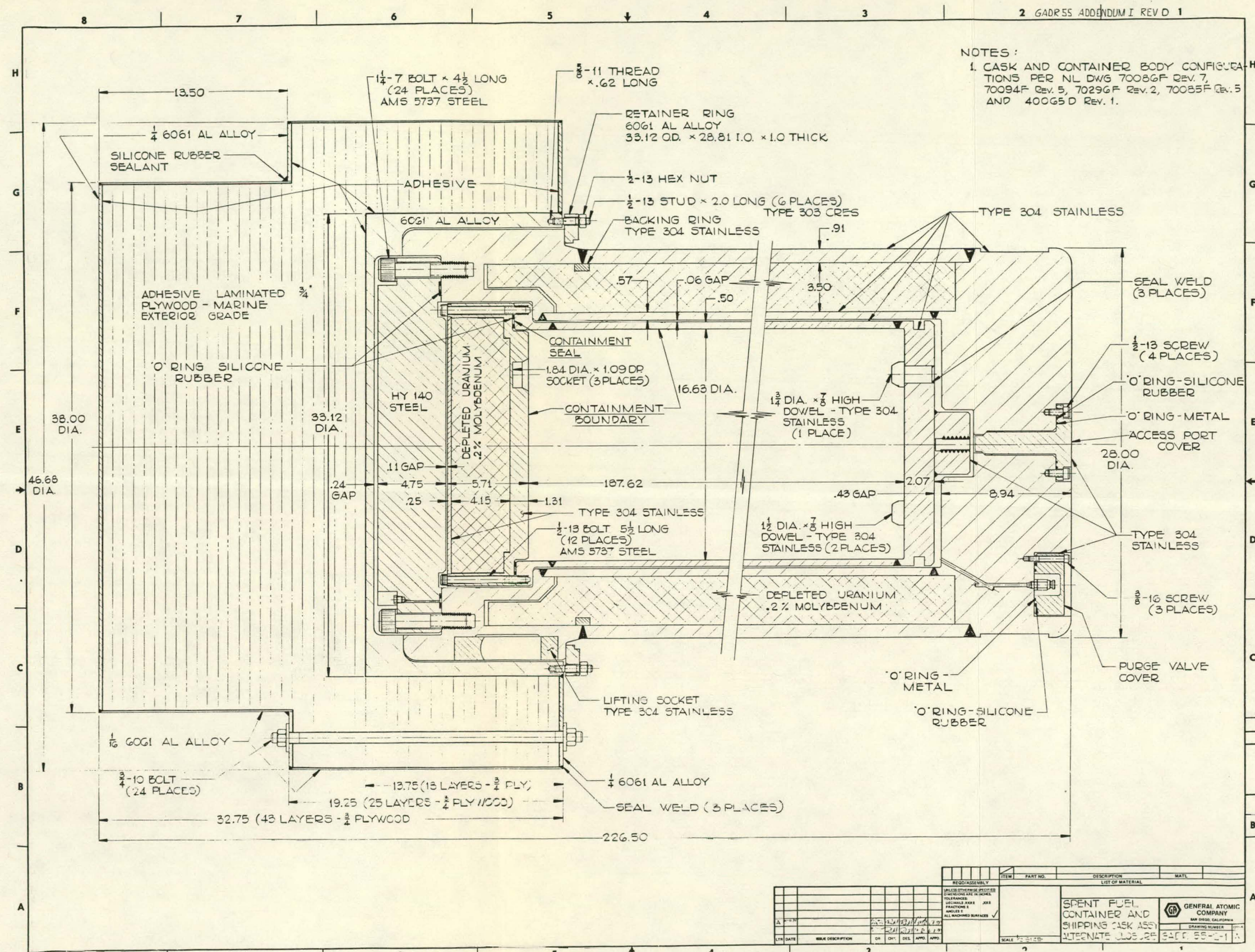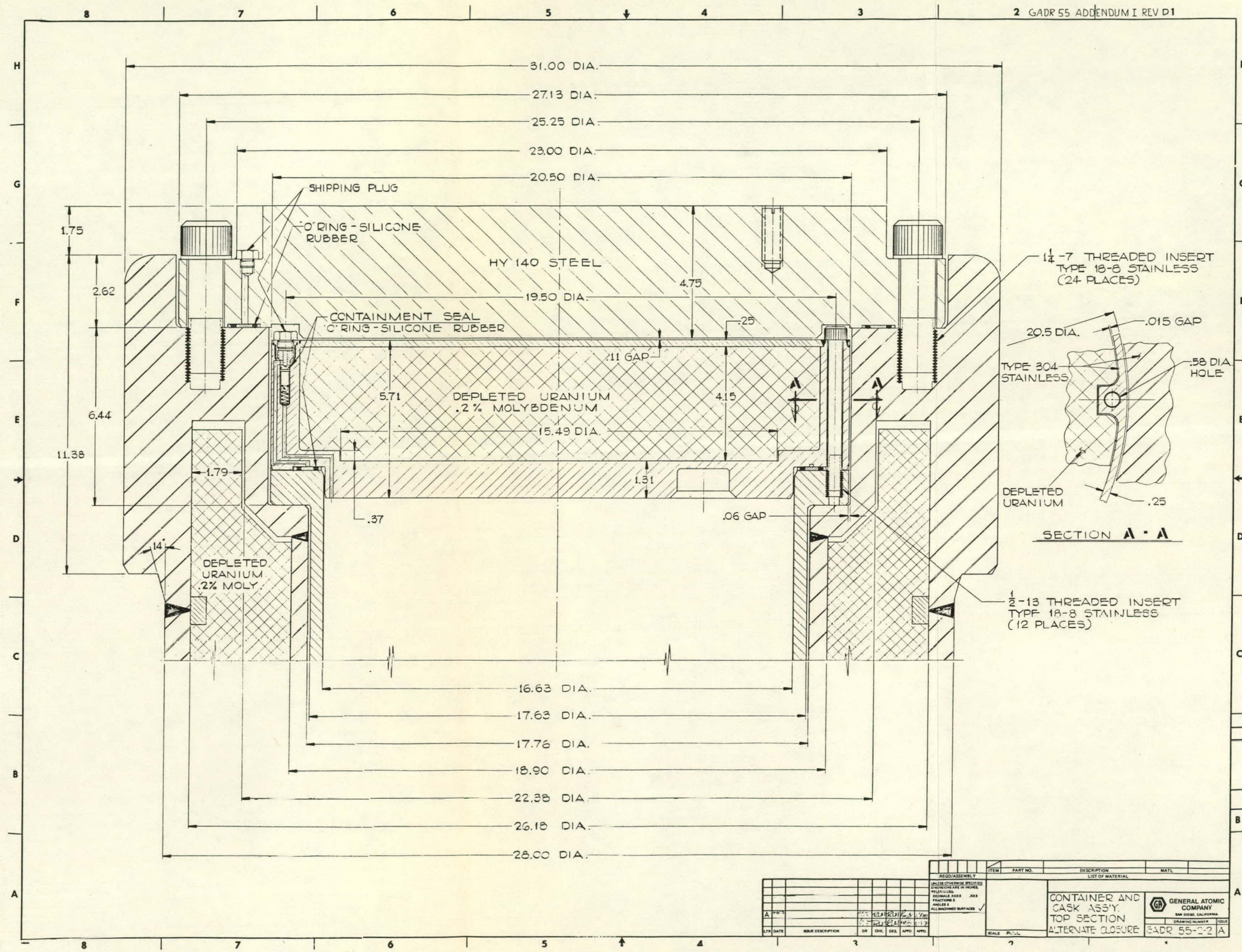SEALANT

ADHESIVE

½-13 HEX NUT

½-13 STUD × 2.0 LONG (6 PLACES)
TYPE 303 CRES

6061 AL ALLOY

BACKING RING
TYPE 304 STAINLESS

.91

TYPE 304 STAINLESS

SEAL WELD
(3 PLACES)

ADHESIVE LAMINATED ¾″
PLYWOOD - MARINE
EXTERIOR GRADE

.57   .06 GAP   3.50
       .50

CONTAINMENT
SEAL

'O' RING SILICONE
RUBBER

HY 140
STEEL

1.84 DIA. × 1.09 DR
SOCKET (3 PLACES)

16.63 DIA.

½-13 SCREW
(4 PLACES)

'O' RING-SILICONE
RUBBER

1¾ DIA. × ⅞ HIGH
DOWEL - TYPE 304
STAINLESS
(1 PLACE)

'O' RING-METAL

CONTAINMENT
BOUNDARY

38.00
DIA.

33.12
DIA.

DEPLETED URANIUM
.2% MOLYBDENUM

ACCESS PORT
COVER

28.00
DIA.

46.68
DIA.

.11 GAP

.24
GAP

4.75

.25

5.71

4.15

1.31

187.62

.43 GAP

2.07

8.94

1½ DIA. × ⅞ HIGH
DOWEL - TYPE 304
STAINLESS (2 PLACES)

TYPE 304
STAINLESS

TYPE 304 STAINLESS

½-13 BOLT 5½ LONG
(12 PLACES)
AMS 5737 STEEL

DEPLETED URANIUM
.2% MOLYBDENUM

⅜-16 SCREW
(3 PLACES)

LIFTING SOCKET
TYPE 304 STAINLESS

'O' RING -
METAL

PURGE VALVE
COVER

'O' RING-SILICONE
RUBBER

1/16 6061 AL ALLOY

¾-10 BOLT
(24 PLACES)

¼ 6061 AL ALLOY

13.75 (18 LAYERS - ¾ PLY)

19.25 (25 LAYERS - ¾ PLY //OOD)

32.75 (43 LAYERS - ¾ PLYWOOD)

SEAL WELD (3 PLACES)

226.50

31.00 DIA.

27.13 DIA.

25.25 DIA.

23.00 DIA.

20.50 DIA.

SHIPPING PLUG

O'RING - SILICONE RUBBER

1.75

2.62

HY 140 STEEL

4.75

1¼-7 THREADED INSERT
TYPE 18-8 STAINLESS
(24 PLACES)

CONTAINMENT SEAL
'O'RING - SILICONE RUBBER

19.50 DIA.

.25

.015 GAP

20.5 DIA.

.11 GAP

TYPE 304
STAINLESS

.58 DIA
HOLE

6.44

5.71

DEPLETED URANIUM
.2 % MOLYBDENUM

4.15

15.49 DIA.

11.38

1.79

1.31

DEPLETED
URANIUM

.25

.37

.06 GAP

SECTION A·A

14°

DEPLETED
URANIUM
.2% MOLY.

½-13 THREADED INSERT
TYPE 18-8 STAINLESS
(12 PLACES)

16.63 DIA.

17.63 DIA.

17.76 DIA.

18.90 DIA.

22.38 DIA.

26.18 DIA.

28.00 DIA.

TO VACUUM
PUMP AND
GAUGE

TO GAS SAMPLING
SYSTEM

¾-16 EXTERNAL, ½-20
INTERNAL THREADED
INSERT TYPE
303 CRES

CASK
COVER

PRIMARY PLUG

'O' RING - TEFLON
PER AMS 3651

CASK
BODY

CONTAINER
LID

CASK
BODY

CONTAINER
LID

CASK
BODY

CONTAINER
LID

.110

18.770 DIA.

CONTAINER
BODY

CONTAINER
BODY

CONTAINER
BODY

16.58 DIA.

SHIPPING CONFIGURATION

SEAL TESTING CONFIGURATION

GAS SAMPLING CONFIGURATION

SEAL - SILICONE
RUBBER

SEAL RETAINER
TYPE 304 STAINLESS

.032 TYP.

.120

.120 TYP.

16.62 I.D.

18.72 O.D.

DETAIL - MAIN FLANGE SEAL FOR
CONTAINER
SCALE 4/1

½-20 THREAD
TYPE 416
STAINLESS

.094 DIA. HOLE

'O' RING SEAL
SILICONE RUBBER

3/16 HEX SOCKET

.69

.373 DIA.

DETAIL - PRIMARY PLUG
SCALE 4/1

3/16 HEX HEAD

'O' RING SEAL
BUNA N RUBBER

.67

½-20 THREAD
TYPE 303 CRES

DETAIL - SHIPPING PLUG
SCALE 4/1

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APPENDIX D

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Bulletin S-82

July 1979

BORO-SILICONE SHIELDING

Type 236 borated silicone is fire-resistant and has a high hydrogen content. It will withstand temperatures up to 400°F (204°C) on a continuous basis.

Type 236 Boro-Silicone has a hydrogen content equivalent to two-thirds that of pure water. In addition, it contains 1 weight-percent boron for capturing thermal neutrons and reducing capture gamma radiation. Although this is a solid material, it is quite resilient, thus minimizing any possible damage in the event of creation of secondary missiles. It is strong enough for handling and self-support purposes. The density of the material is 1.6 g/cc. Type 236 Boro-Silicone is a self-extinguishing material. It is available in a variety of shapes and sizes.

TECHNICAL DATA

Properties

Hydrogen: 4.4×10^{22} atoms/cc

Boron: 9.0×10^{20} atoms/cc

Weight-Percent Boron: 1.0%

Macroscopic Thermal Neutron Cross Section, $\Sigma = 0.68$ cm^{-1}

Density: 1.6 g/cc = 100 lbs/cu ft

Recommended Temperature Limit: 400°F = 205°C

Machinability: Good

Hardness: Shore "A" Durometer Scale = 60

Flammability (ASTM D635): Self-extinguishing with glowing
combustion. Average time to self-extinguish = 0 seconds.
Average extent of burning = 0.2".

Coefficient of Thermal Conductivity: 5.8×10^{-3} cal-cm/sec cm^2 °C
= 1.4 BTU-ft/hr ft^2 °F

Heat Capacity (Specific Heat): 0.4 cal/g°C

Cubical Coefficient of Expansion: 5.2×10^{-4} cc/cc°C
= 3×10^{-4} cu in/cu in°F

Linear Coefficient of Expansion: 1.7×10^{-4} cm/cm°C
= 1×10^{-4} in/in°F

Tensile Strength (ASTM D638): 59 psi

Radiation Resistance, gammas: 2×10^8 rads

Radiation Resistance, neutrons: 5×10^{17} n/cm^2

EQUIPMENT FOR ■ REACTORS ■ ACCELERATORS ■ LABORATORIES

GENERAL ATOMIC COMPANY
P. O. BOX 81608
SAN DIEGO, CALIFORNIA 92138