SECTION 3.0

THERMAL EVALUATION

3.0 THERMAL EVALUATION

A separate thermal evaluation for Model FSV-1 in Configurations F and G is not provided since thermal evaluation for Model FSV-1 in Configuration E will encompass all applicable thermal conditions.

3.1 DISCUSSION

Model FSV-1 in Configuration E is designed for the transport of six (6) High Temperature Gas-Cooled Reactor spent fuel elements. Each spent fuel element has a maximum decay heat output of 2322 Btu/hr after 100 days following reactor shutdown.

3.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Table 3-1 lists the thermal properties of the materials which constitute the cask as they were used in the analysis. In the formulas shown, T represents the temperature in °R and t in °F.

TABLE 3-1
THERMAL PROPERTIES OF MATERIALS

Helium Conductivity

$K = 0.00129\ T^{0.674}$ Btu/hr-ft°F) Ref. 3-9

Air Conductivity

$K = 0.0146 + 1.695\ E\text{-}05\ t$ Btu/hr-ft°F) Ref. 3-10

Helium-Air Mixture (50% by Volume) Conductivity

$K = 0.341 \bullet K_{He} + 0.549\ K_{Air}$ Ref. 3-11

Stainless Steel (Type 304)

Conductivity

$K = 8.0 - 0.004433\ t$ Btu/hr-ft°F) Ref. 3-12

Heat Capacity per Unit Volume

$C = 55. + 0.011458\ t$ Btu/ft^3-°F) Ref. 3-12

Emissivity

$= 0.85$ Ref. 3-8

Aluminum (Type 6061)

Conductivity

$K = 100$ Btu/hr-ft°F) Ref. 3-13

Heat Capacity per Unit Volume

$C = 39.$ (Btu/ft^3-°F) Ref. 3-13

Emissivity

$\epsilon = 0.2$ Ref. 3-7

$\epsilon = 0.85$ (with surface treatment for exposed surface)

TABLE 3-1 (continued)

Depleted Uranium Shielding Material	
Conductivity	
K = 14.8 Btu/hr-ft°F)	Ref. 3-2
Heat Capacity per Unit Volume	
C = 38. (Btu/ft^3-°F)	Ref. 3-2
Emissivity	
ϵ = 0.5	Ref. 3-2
Spent Fuel Blocks	
Conductivity	
K = 10.0 Btu/hr-ft°F)	Ref. 3-2
Heat Capacity per Unit Volume	
C = 32. (Btu/ft^3-hr)	Ref. 3-2
Emissivity	
ϵ = 0.8	Ref. 3-2
Alloy Steel (4340)	
Conductivity	
K = 20.0 Btu/hr-ft°F)	Ref. 3-12
Heat Capactiy per Unit Volume	
C = 60. (Btu/ft^3-hr)	Ref. 3-12
Emissivity	
ϵ = 0.1	Ref. 3-7

3.3. TECHNICAL SPECIFICATIONS OF COMPONENTS

A Gask-O-Seal assembly is used to seal the inner closure to the inner container body and an O-ring is used to seal the cavity gas sampling port. Both of these seals are manufactured from a silicone elastomer which has a useful temperature operating range from -80°F to +450°F and will resist temperatures up to +700°F for short periods of time.

3.4. THERMAL EVALUATION FOR MODEL FSV-1 CONFIGURATION E

The following evaluation combines the package performance during the normal conditions of transport and the hypothetical accident conditions.

3.4.1. Thermal Model

Temperatures were calculated by a finite-difference method. A digital computer code, TAC2D described in Ref. 3-2, was used to perform the numerical calculations concerning the upper half of the cask. The lower end was originally analyzed with the aid of RAT. The cask system was modeled in a form suitable for both of these codes.

TAC2D is a digital computer code that is capable of calculating temperature transients for a two-dimensional network of points. It may also be used to obtain steady state solutions asymptotically by carrying a transient calculation to the point where the time dependence of the result becomes negligible. The code version used for the present analysis differs only in array dimension statements from a certified version evaluated in Ref. 3-3. All other program adaptations relate to the particular problem at hand and are documented with the actual computer runs in Ref. 3-4.

The network is specified by establishing a grid system, assigning individual materials within that grid system, and identifying the applicable thermal parameters that represent those materials. The grid system consists of two sets of grid lines parallel to the axes in a cylindrical coordinate system.

Material blocks are defined by four bounding grid lines and a label which identifies the material they contain. Parameters that characterize each material are the pertinent thermal properties as well as the volumetric rate of heat generation within the material. Material blocks may be separated by narrow gaps that contain stagnant gases which are identified by numbers allocating the proper thermal conductivities. Heat transfer across these gaps is one-dimensional by conduction and radiation.

The various material regions and gap configurations are illustrated on Figs. 3-1 and 3-2.

Boundary conditions at external boundaries are imposed either by prescription of a sink temperature and surface conductance or by a simulated flowing coolant of given properties and flow conditions. In the present case, the interface between the plywood and the metal support structure of the impact limiter was assumed to constitute in effect an adiabatic boundary. Through the remaining cask surfaces, heat is exchanged with the surroundings by convection, radiation and, when applicable, as a prescribed heat flux due to solar irradiation.

The thermal parameters for the materials, gases in the gaps and coolants have been incorporated in the computer code in functional form. Certain dependent variables are available for use in these functions thereby permitting a nonlinear representation of the problem.

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 3-1. Gap Configuration

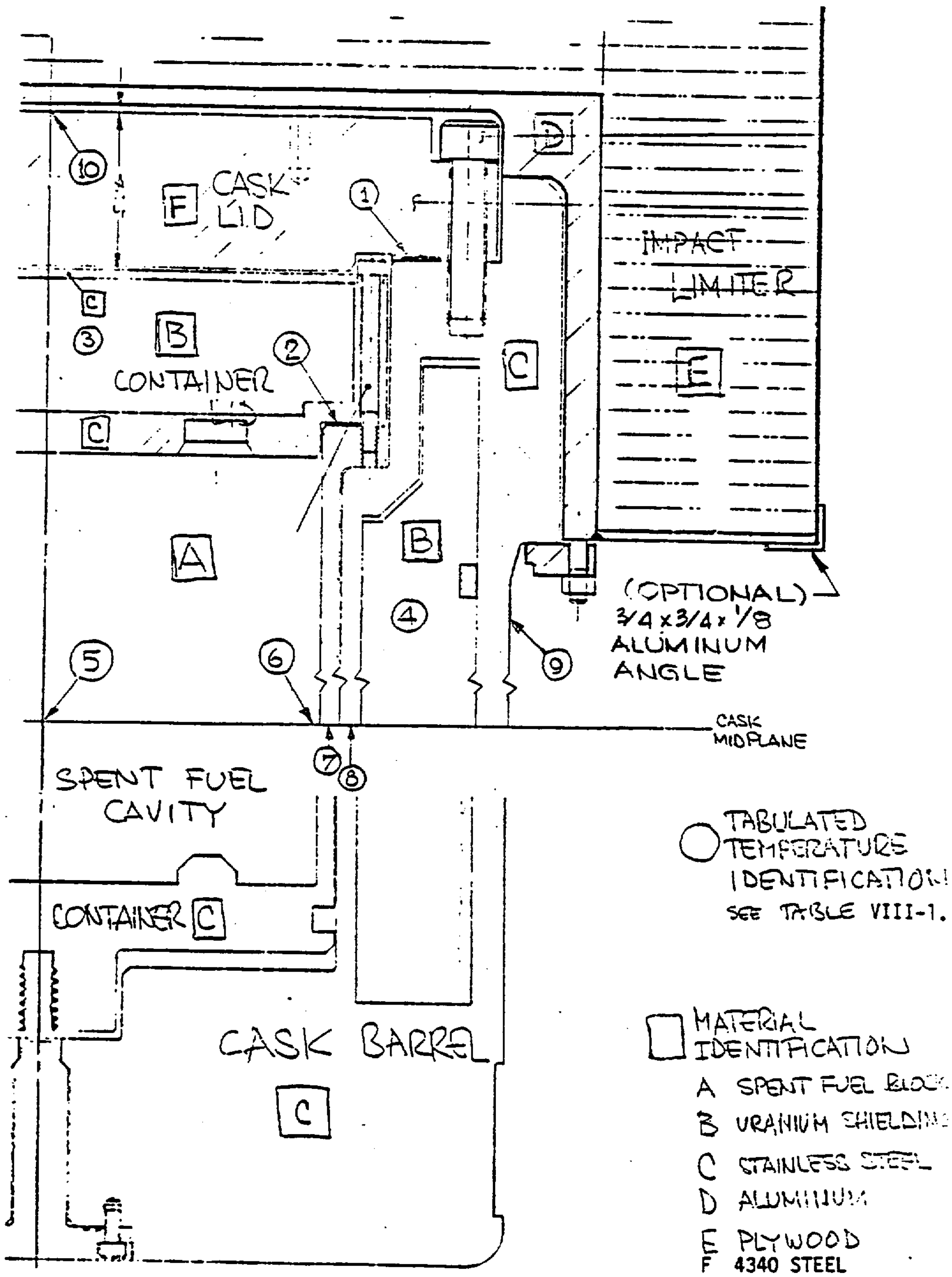

Figure 3-2. Shipping cask model

The outer closure and inner closure seal assemblies contain material of limited lifetime where this lifetime is temperature dependent. The probability that the seal fails at any time during the entire transient is obtained as the sum of the interval failure probabilities. The functional relationship between seal life and operating temperature is illustrated on Fig. 3-3, which is based on Ref. 3-5.

3.4.2. Criteria for Evaluation

The following three situations were used to define the boundary and initial conditions for the analyses:

Condition 1A. Maximum temperature day. This condition assumes an ambient temperature of 130°F around all parts of the cask. Radiation and natural convection are the mechanisms by which heat is removed from the cask. Solar heating is imposed with an effective intensity of 96 Btu/h-ft^2 on the cylindrical portion of the exposed cask surface and half of it on the exposed vertical surfaces. Decay heat generation rates for spent fuel conditions 100 days after reactor shutdown were used.

Condition 1B. Same as condition 1A except that spent fuel conditions 200 days after reactor shutdown were used.

Condition 2. Minimum temperature day. This refers to a condition of -40°F ambient temperature without solar heating. The 200 day heat generation was used for comparison with Condition 1B.

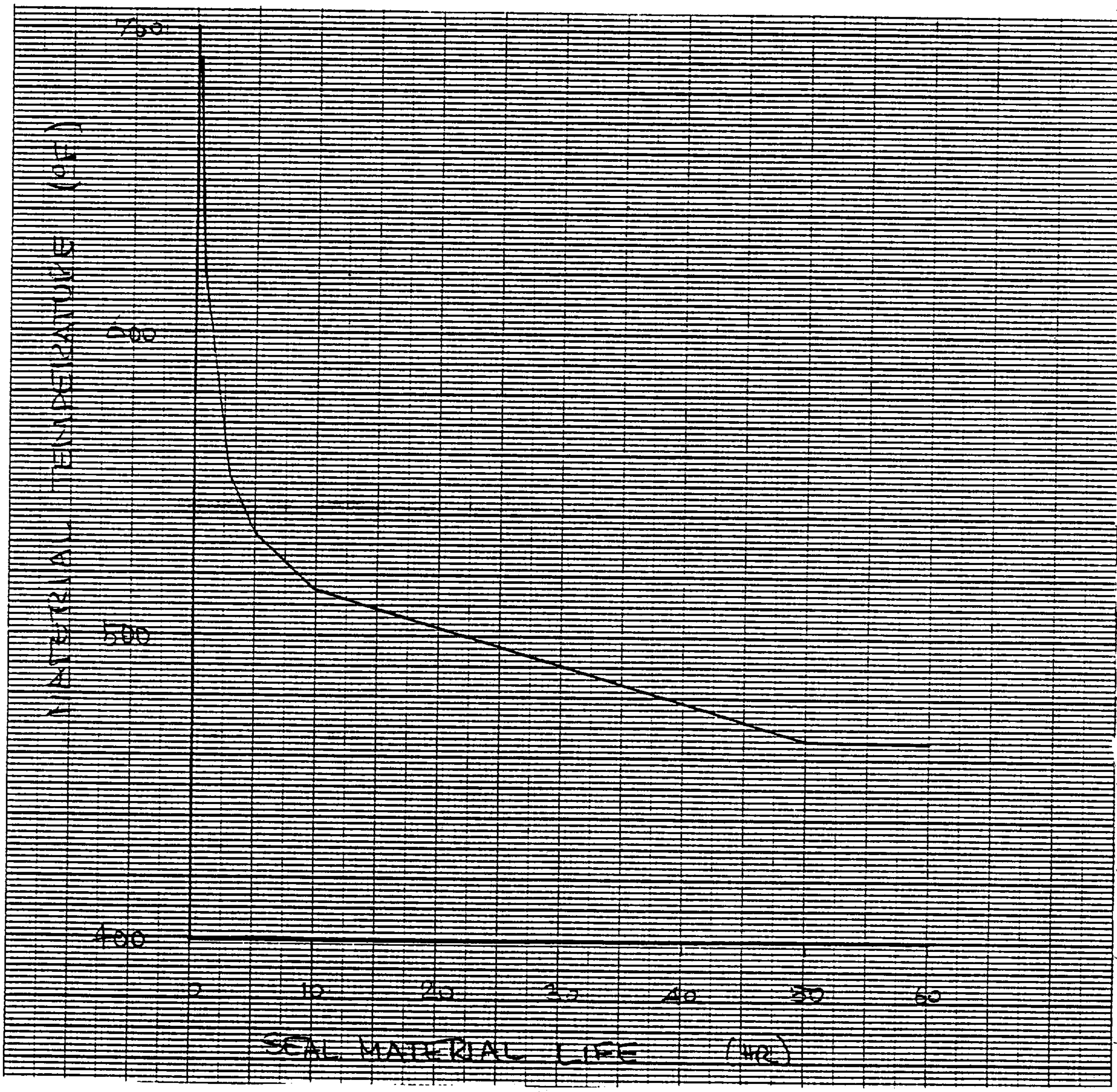

Figure 3-3. Seal material temperature limitations

Condition 3. Fire accident. This accident assumes a 1475°F fire completely surrounding the cask for a period of 30 minutes on the maximum temperature day. A convection heat transfer coefficient of 300 Btu/h-ft² °F was added during the fire to simulate a severe convection condition due to hot, blowing gases. After the fire, the cask is returned to the 130°F day.

3.4.3 Thermal Evaluation

a. Maximum Temperature Day: The maximum temperature day is defined as sunny and having an air and radiation background temperature of 130°F. The net solar irradiation is assumed to be constant and amount of 96 Btu/h-ft² on the cylindrical portion of the exposed cask and impact limiter. (This assumption permits the calculation of a representative steady-state solution in lieu of a temperature cycle for a 24-h period.) Further, it was assumed that the cask is surrounded by still air so that convective cooling is limited to free convection. Using a correlation for free convection around a horizontal cylinder, the following function was derived to calculate the surface heat transfer coefficient (Ref. 3-7):

$$h = 0.221 \left(S_t - T_a\right)^{0.25}$$

where S_t = the cask surface temperature,
T_a = the ambient temperature.

Thermal radiation from the cask to its surroundings was included utilizing a cask surface emissivity of 0.85 (Ref. 3-8) and a background absorbtivity of 1.0.

b. Minimum Temperature Day: The minimum temperature day was defined as having an air radiation background temperature of -40°F and no solar heating. This heat exchange between the cask and its surroundings is calculated on the same basis as that of the maximum temperature day.

c. Fire Accident: The fire accident is defined as a temporary radiation source of 1475°F whose iteraction with the cask is based on a fire emissivity of 0.9 and a cask surface absorptivity of 0.8. An additional surface heat transfer contribution resulting from strong convection was assumed. Heat transfer coefficients representing hot, flowing gases are expected to reach at most 300 Btu/h-ft^2-°F. The impact limiter, largely made from plywood, protects the cask from the fire. Its plywood bulk is assumed to perfectly insulate the major portion of the structural aluminum elements. There is no heat input due to solar irradiation during the fire.

Cask Dimensions:

Since the thermal analysis is performed with cask dimensions determined from manufacturing drawings, gaps cannot be defined exactly. Their actual size will fall into a range bounded by extremes which can be deduced from specific manufacturing tolerances. Sets of extreme combinations (such as all gaps of the largest possible size) as well as a mixed set which tends to affect the seal temperatures in an adverse manner have been utilized in the calculation of the results presented here. Thermal expansion of materials and the associated variation in gap sizes were considered insignificant and therefore disregarded in the analysis.

Spent Fuel Heat Generation:

The fuel blocks to be shipped in this cask will have been irradiated and the original fissionable material partly consumed. Due to residual isotope activity in blocks that have been removed from the reactor, there is a continuing "after-heat" generation which decreases with time. It takes place predominantly in the fuel and graphite blocks and to a lesser degree in cask shielding components. This heat generation is predictable and has been used in the calculations.

It is assumed that the spent fuel is loaded into the shipping cask no sooner than 100 days after the reactor has been shut down. This represents the maximum heat generation that the cask need be designed for.

The fuel element behavior after reactor shutdown and the associated after-heat generation were evaluated at GA in unrelated analyses. The heat generation rates recommended for thermal analysis of short duration processes are listed in Table 3-1. Of these quantities, 88% are realized within the fuel block and the remaining 12% are generated within the first inch of the surrounding shielding. For the present analysis, all heat generation was assumed to be confined to the fuel blocks and distributed uniformly through their entire volume.

3.4.4 Results of Thermal Evaluation

Steady-state and transient thermal analyses were conducted with the aid of two models representing each end of the cask extending far enough along the length of the cask to ascertain valid results. No unexpected or in any way critical results were obtained from the analysis of the relatively simple lower end. Results pertaining to the upper end of the cask including the effect of the impact limiter are summarized in Table 3-2. The analytical model of the cask's thermal behavior is described in the following sections.

Three gap combinations have been constructed and employed in this analysis. The minimum gap and maximum gap combinations refer to extreme geometrical situations whereas the mixed gap combination refers to a set of gaps which was considered likely to cause high temperatures in the top of the cask under the insulating impact limiter where the critical seal assemblies are located. Temperature histories of two fire accident transients are shown on Fig. 3-4a and Fig. 3-4b, where the corresponding calculated seal life reduction due to the fire is also listed. Based on the similarity of these two results, it was concluded that other gap combinations would yield temperatures within the basic computational tolerances of those reported. Figures 3-5 through 3-8 show the complete temperature maps from the computer output for representative steady-state conditions for the extreme gap configurations.

3.4.5 Conclusions

The calculated temperatures, as provided in Table 3-2, for the three conditions evaluated do not exceed any design parameters or acceptable operating temperatures.

TABLE 3-2
TEMPERATURES AT SELECTED LOCATIONS

	Condition 1A			Condition 1B	Condition 2		Condition 3		
	130°F Ambient Air Solar Irradiation				-40°F Ambient Air		Accident: Fire of 30 min duration		
	100 Day Fuel*			200 Day Fuel**	200 Day Fuel		(Peak Values)		
	Min Gaps	Max Gaps	Mixed Gaps	Max Gaps	Min Gaps	Max Gaps	Min Gaps	Max Gaps	Mixed Gaps
1) Cask Seal	242	234	244	202	30	23	517	463	522
2) Container Seal	270	274	284	224	50	51	484	454	485
3) Closure Shielding	282	274	299	224	59	51	482	452	486
4) Body Shielding	266	275	274	227	36	46	839	831	830
5) Fuel Centerline	392	422	420	311	140	170	562	574	572
6) Fuel Surface	384	414	412	307	137	166	555	567	565
7) Container Wall	327	362	361	275	80	115	587	581	579
8) Inner Shell	296	324	322	253	57	83	656	646	645
9) Exposed Cask Surface	227	227	228	203	6	7	1440	1442	1442
10) Cask Top Centerline	247	236	247	203	33	24	481	431	487

*Equivalent Heat Generation: 2322 Btu/h/spent fuel element
**Equivalent Heat Generation: 1101 Btu/h/spent fuel element

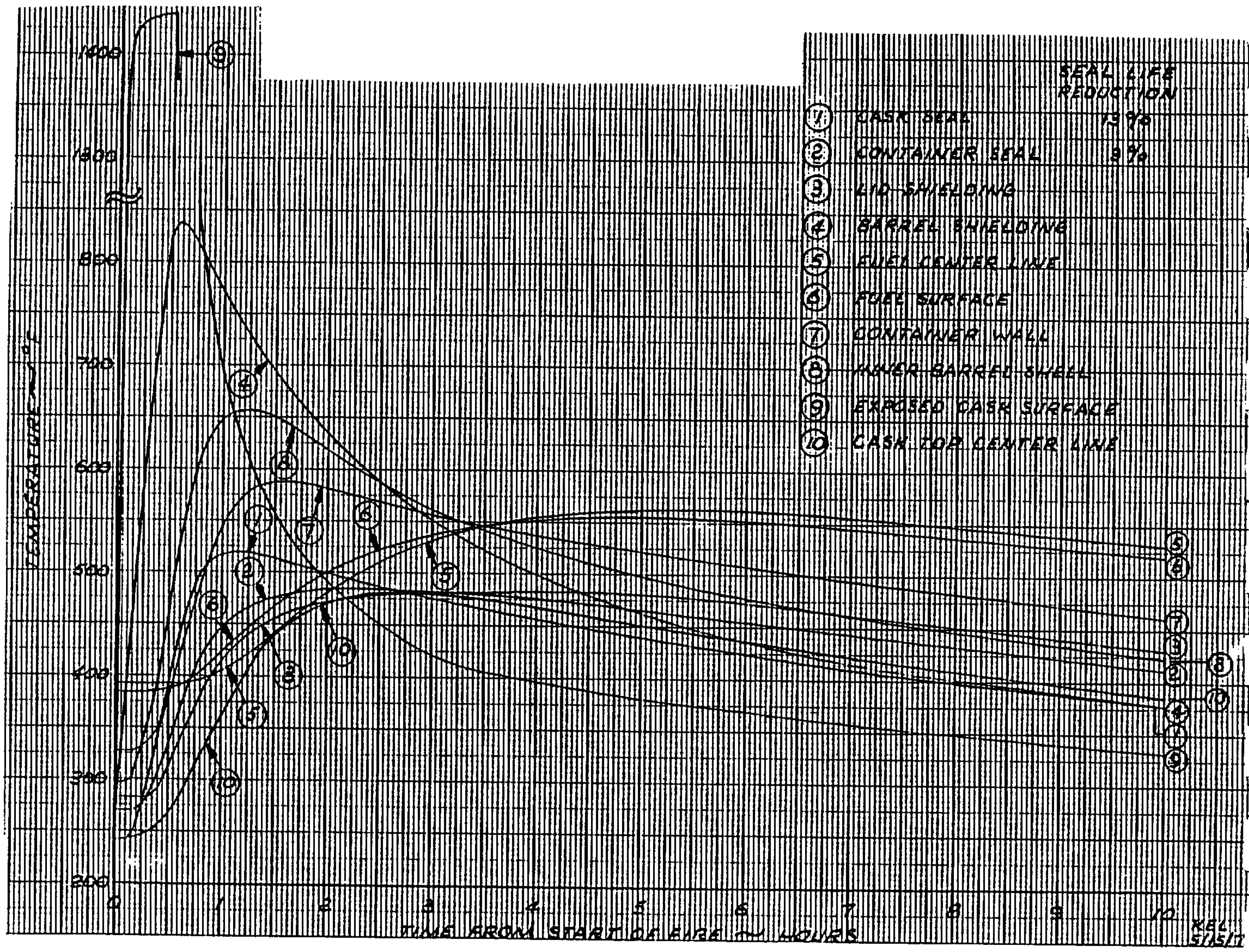

Figure 3-4a. Fire Accident (Minimum Gaps)

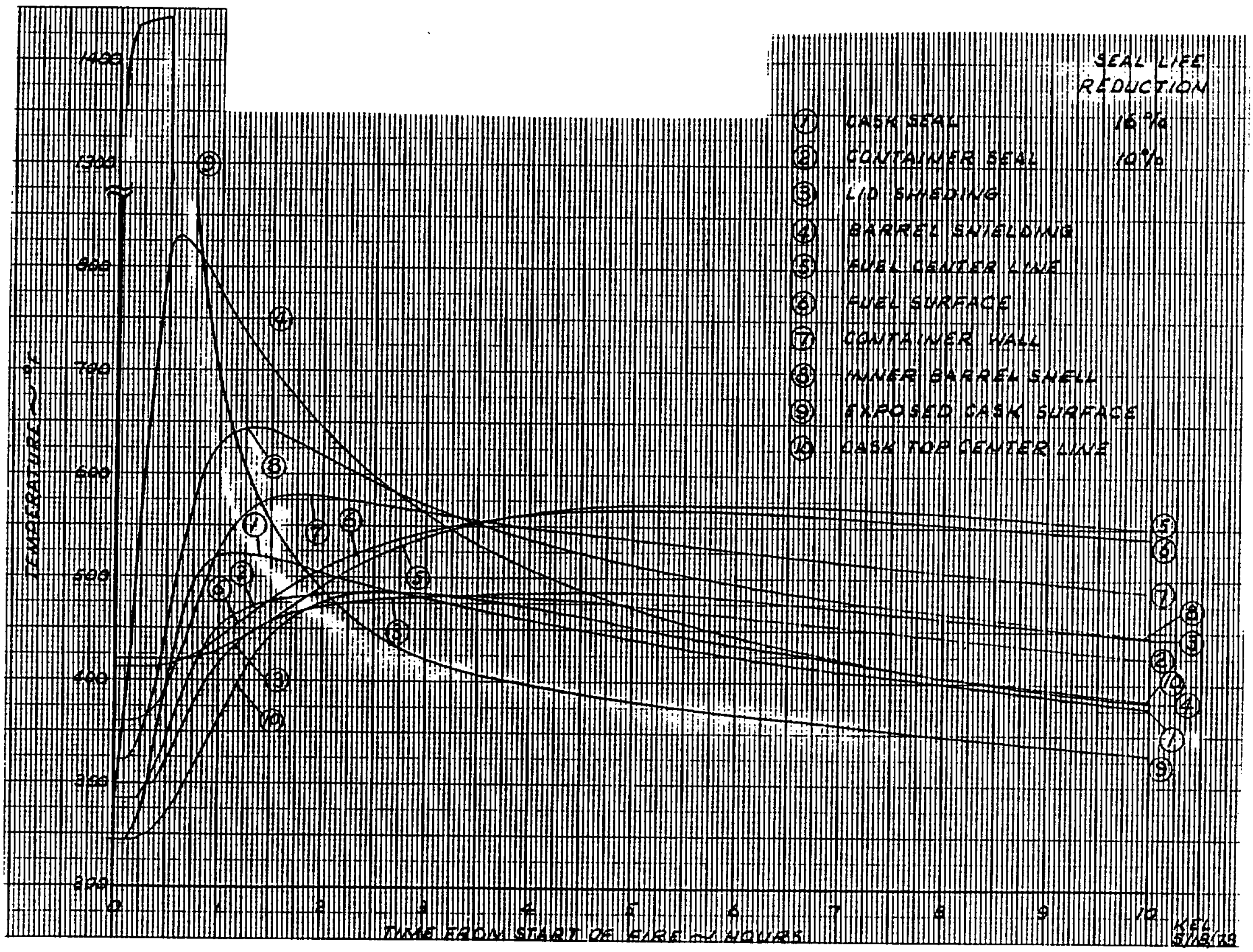

Figure 3-4b. Fire Accident (Mixed Gaps)

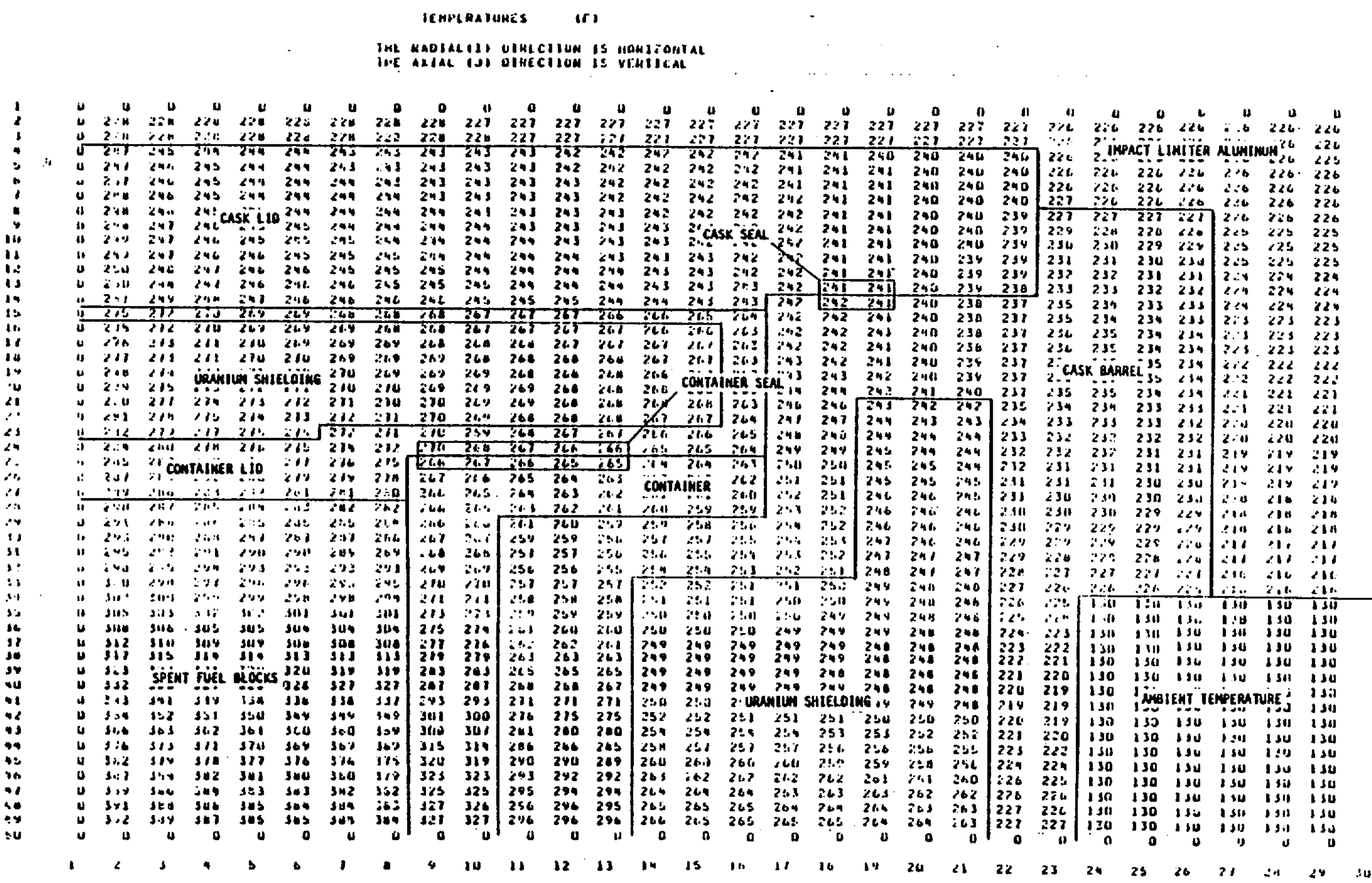

Figure 3-5. Fuel Shipping Cask Temperature Map for Steady State Condition 1A (Maximum Temperature Day) - Minimum Gap Configuration, 100 Day Fuel

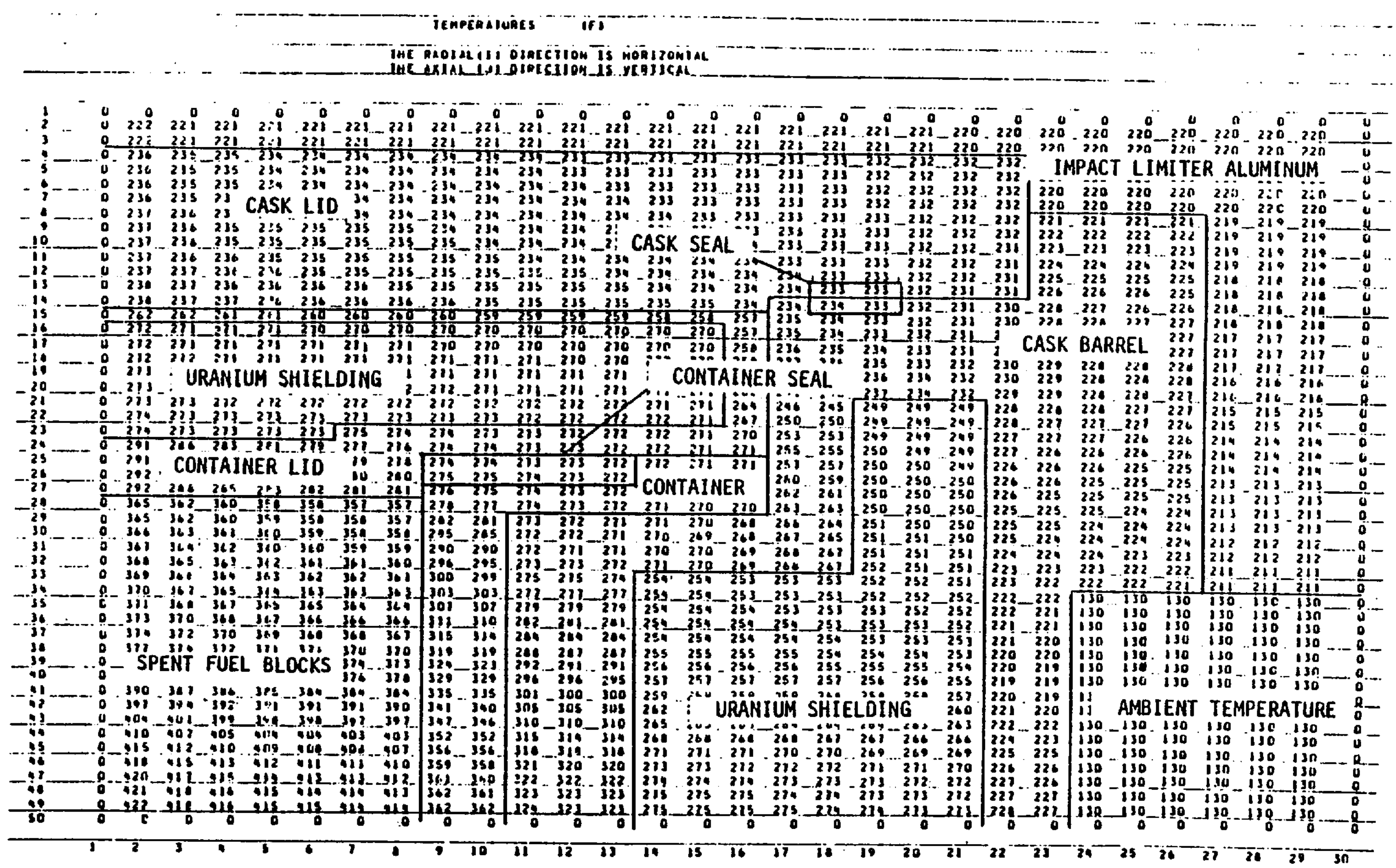

Figure 3-6. Fuel Shipping Cask Temperature Map for Steady State Condition 1A (Maximum Temperature Day) - Maximum Gap Configuration, 100 Day Fuel.

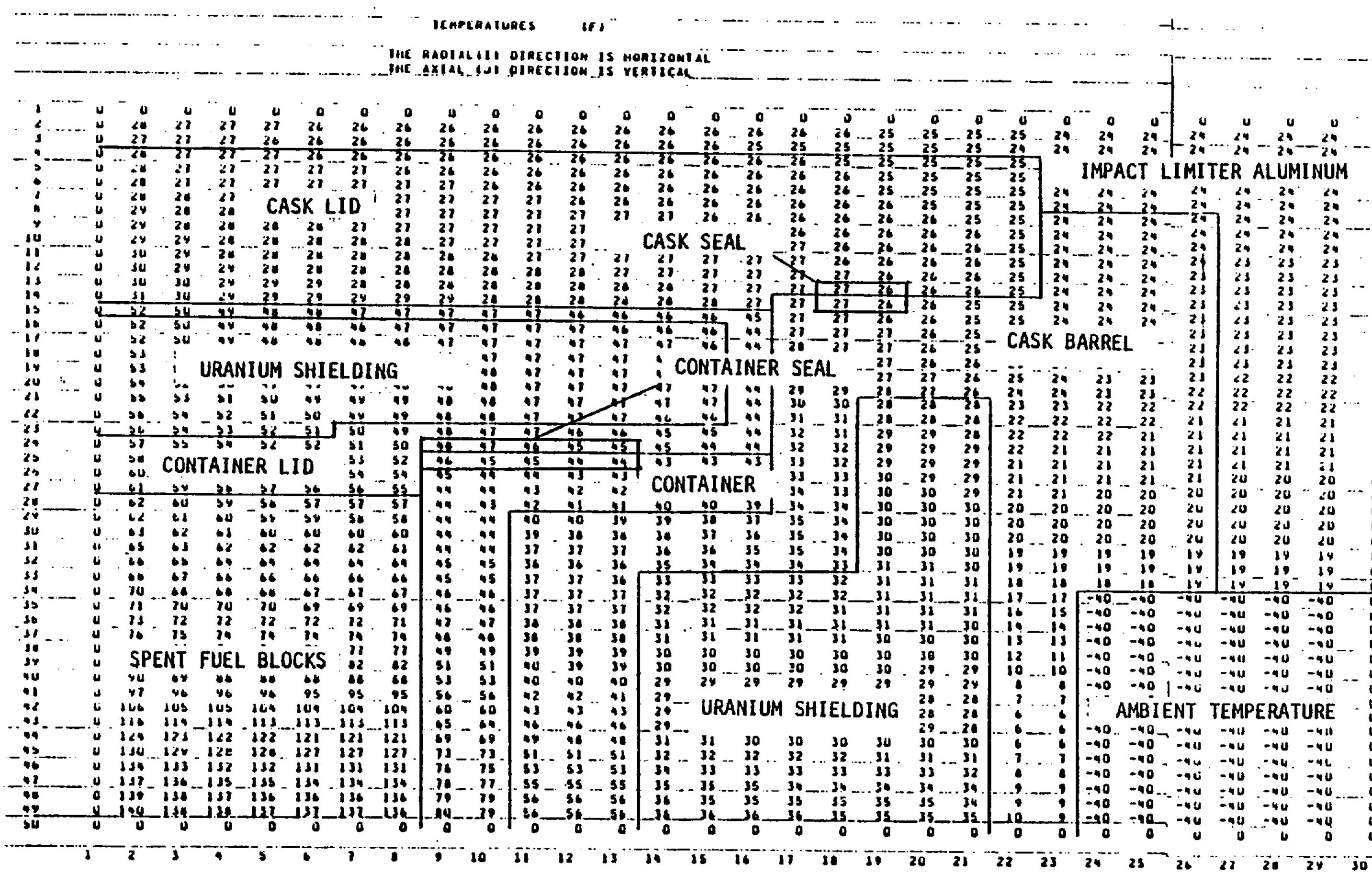

Figure 3-7. Fuel Shipping Cask Temperature Map for Steady State Condition 2 (Minimum Temperature Day) - Minimum Gap Configuration, 200 Day Fuel.

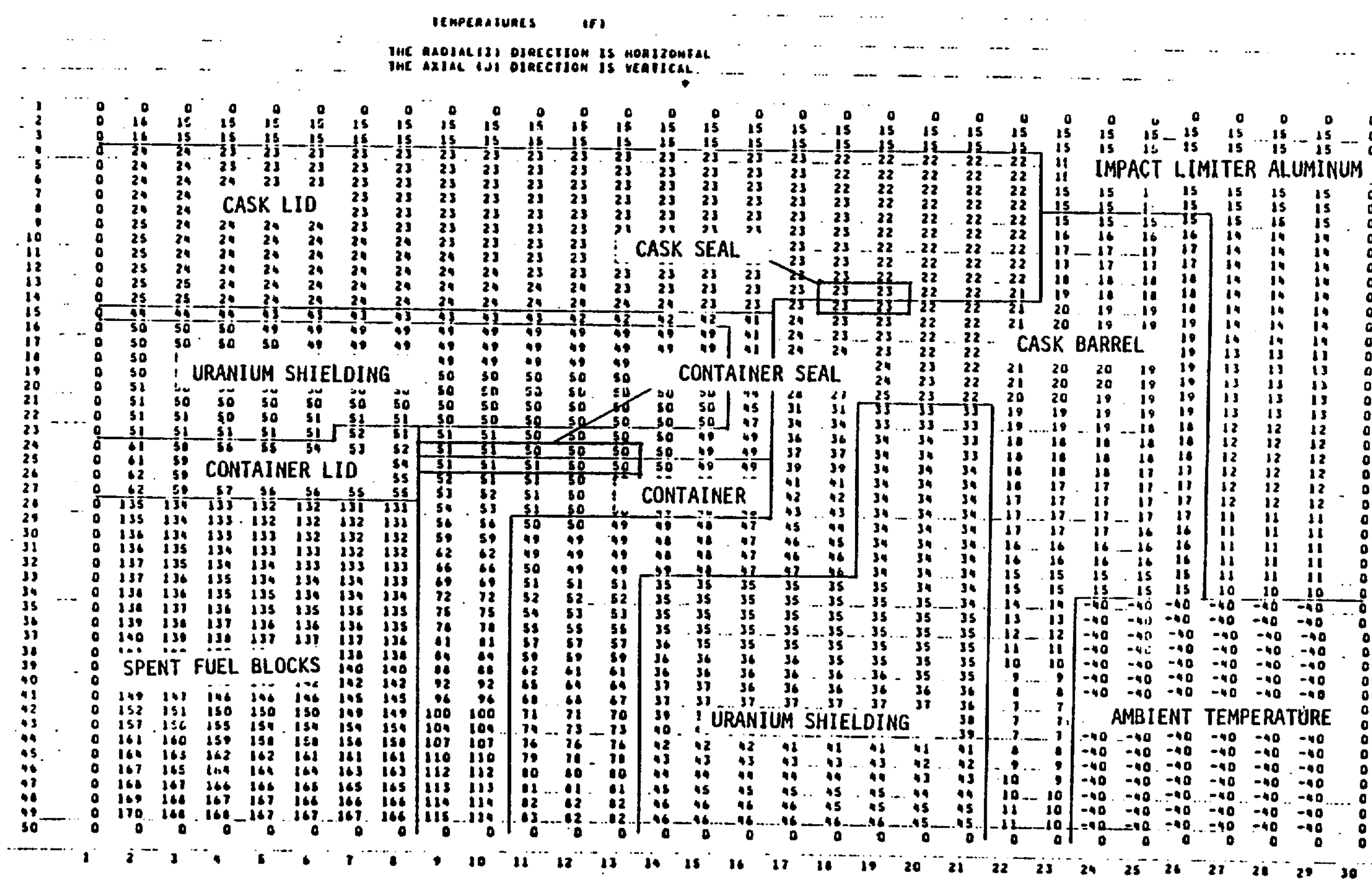

Figure 3-8. Fuel Shipping Cask Temperature Map for Steady State Condition 2 (Minimum Temperature Day) - Maximum Gap Configuration, 200 Day Fuel.

REFERENCES

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SECTION 4.0

CONTAINMENT

4.0 CONTAINMENT

Section 4.1 contains the containment evaluation for Model FSV-1 in Configuration E, when used for the transport of spent fuel elements from a high temperature gas cooled reactor.

The containment evaluation for Model FSV-1 in Configurations F and G when used for the transport of solid nonfissile irradiated and contaminated hardware is provided in Section 4.2 and subsections thereto.

4.1 MODEL FSV-1 CONFIGURATION E

4.1.1. Containment Boundary

4.1.1.1. Containment Vessel

Model FSV-1 in Configuration E consists of the cask body with the outer closure and the inner container with the inner closure. This inner container with associated elastomer seals is the containment vessel. Gamma shielding and structural strength are provided by the cask body.

4.1.1.2. Containment Penetrations

A cavity gas sampling port is located in the closure for the inner container. This port is sealed by an elastomer O-ring which is tested as part of the assembly verification test conducted prior to each use of the package.

4.1.1.3. Seals and Welds

The inner container body consists of the flange, the cylindrical section and the base. These three pieces are joined together with full penetration welds. A Gask-O-Seal assembly manufactured by the Parker Hannifin Corporation is used between the inner container body and the inner closure to prevent the release of radioactive material during the normal conditions of transport and limit the release of radioactive material during hypothetical accident conditions.

4.1.1.4. Closure

The inner closure consists of a stainless steel shell containing depleted uranium gamma shielding that is 4.15 inches thick. Twelve socket head cap screws attach the closure to the inner container body. A cavity gas sampling and seal test port is located in the closure.

4.1.2. Requirements for Normal Conditions of Transport

The radioactive contents of Model FSV-1 in Configuration E are six (6) spent fuel elements from the Fort St. Vrain High Temperature, Gas-Cooled Reactor. These fuel elements consist of graphite blocks containing fuel rods and coolant passages. The fuel rods contain the nuclear fuel in the form of micro-spheres of either uranium which has been enriched to 93% U-235 or thorium as fertile material to produce U-233. These micro-spheres are sealed within a silicon carbide shell which contains the nuclear fuel and any fission products generated during fuel burn up in the reactor. In case of any failures of these silicon carbide shell during reactor operation the fission products will mix with the helium coolant and then be removed by the coolant purification system. Therefore during the normal conditions of transport there is no radioactive material available for release from Model FSV-1 in Configuration E.

4.1.3. Containment Requirements for the Hypothetical Accident Conditions

4.1.3.1. Fission Gas Products

The maximum fission product activity that will be contained in Model FSV-1 in Configuration E at any time would be that associated with six elements having the full 6-year burnup. Total fission product per fuel element is given in Table 4-1.

Assuming an average power peaking factor of 1.35 over the 6 years in the reactor and 100 days of decay since shutdown, the volatile (noble gases and iodine) inventory in the fuel elements would amount of about 4700 curies of noble gas (principally Kr-85) and 23 curies of iodine (principally I-131). In addition, about 9 gm moles of stable noble gas and iodine would be present. The total amount of volatile fission product gases would occupy a volume of about 7.5 ft^3 STP.

This fission product activity is doubly contained, first by the fuel particles and their impervious TRISO coatings, and secondly by the inner container. Fission products can be annealed from fuel particles, but not in significant quantity at temperatures below 1000°F.

If a breached cask with a failed inner canister is assumed to be immersed in water, then the potential exists for hydrolysis of the carbide kernels of fuel particles with previously failed coatings. With unrestricted exposure to moisture, essentially complete hydrolysis of all failed fuel would be expected with a corresponding release of the noble gas

TABLE 4-1
FUEL ELEMENT FISSION PRODUCT INVENTORY

Equilibrium Core, 6-Year Operation, 100 Days Shutdown

Isotope	Activity (Curies)
Kr-85	786.
Sr-89	8600.
Sr-90	2300.
Y-90	2300.
Y-91	11900.
Zr-95	13600.
Nb-95m	270.
Nb-95	13600.
Ru-103	2700.
Rh-103m	2700.
Ru-106	1600.
Te-127m	270.
Te-127	270.
Te-129m	390.
I-131	24.
Te-129	390.
Cs-137	2250.
Ba-137m	2080.
Ba-140	186.
La-140	186.
Ce-141	5100.
Pr-143	260.
Ce-144	25200.
Pr-144	25200.
Pm-147	9500.
Sm-151	470.
Pa-233	34000.

activity, i.e., 100% release for 100% hydrolysis. No iodine has been observed to be released by this mechanism. If the upper design limit for the failed fuel particle fraction of 10% is assumed, along with complete hydrolysis of this fraction, then the noble gas release would amount to about 470 curies of Kr-85. The potential dose from this activity is dependent on the atmospheric conditions and the distance between the source and receptor which combine to make up the effective atmospheric dilution factor. Conservatively assuming a dilution factor of 0.1 sec/m^3 and that the receptor is immersed in a semi-infinite cloud of released activity, then the received dose would be only about 2.5 millirem according to the following calculation:

$$\text{Dose (rem)} = 0.247 \times \text{curies released} \times \text{gamma energy}$$

$$(\text{mev/dis.}) \times \text{dilution factor (sec/m}^3)$$

$$= 0.247 \times 470 \times 0.0021 \times 0.01 = 2.45 \times 10^{-3} \text{ rem}$$

The total amount of volatile activity in the cask (4723 curies) and the maximum potential amount releasable (470 curies of Kr-85) are well below the permissible limits.

4.1.3.2. Release of Contents

The limit for the release of radioactive material during the hypothetical accident conditions is determined from NRC Regulatory Guide 7.4 - 1975 and American National Standards Institute (ANSI) standard, ANSI N14.5 -- 1977. For a type of B(U) package this limit is given as $A_2 \times 10^{-3}$ curies per week or $1.65\ A_2 \times 10^{-9}$ curies per second. ANSI N14.5 - 1977 provides the following formulas for determining the corresponding allowable leakage rate as follows:

$$L_A = \frac{R_A}{C_A} \text{ where,}$$

L_A is the leakage rate in cm^3 per second,

R_A is the release rate in curies per second,

C_A is the specific activity in the package in curies per cm^3

Krypton-85 has an A_2 value of 1000 curies and thus the R_A quantity is:

$$R_A = \frac{1000 \times 10^{-3} \text{ curies}}{604{,}800 \text{ sec}} = 1.65 \times 10^{-6} \frac{\text{Ci}}{\text{sec}}$$

Krypton-85 in the amount of up to 470 curies could be available for release and the net void volume in the model FSV-1A package is 235×10^3 cm^3. Therefore, C_A is:

$$C_A = \frac{470 \text{ ci}}{235 \times 10^3 \text{ cm}^3} = 2.0 \times 10^{-3} \frac{\text{ci}}{\text{cm}^3}$$

The maximum permissible leakage rate for hypothetical accident conditions is now calculated as shown below:

$$L_A = \frac{1.65 \times 10^{-6} \text{ ci/sec}}{2.0 \times 10^{-3} \text{ ci/sec}} = \underline{8.25 \times 10^{-4}} \text{ cm}^3\text{/sec}$$

4.2 MODEL FSV-1 CONFIGURATIONS F AND G

4.2.1. Containment Boundary

4.2.1.1. Containment Vessel

The inner container is not a part of Model FSV-1 while in Configurations F or G and therefore the cask body and the outer closure with the associated seal assembly constitute the containment vessel.

4.2.1.2. Containment Penetrations

There are two penetrations located in the base of the cask body. Each of these penetrations has a bolted closure with an elastomer and a metal O-ring.

4.2.1.3. Seals and Welds

The inner shell of the cask body is joined to the base section of the cask and to the closure section at the other end with full penetration welds.

A Gask-O-Seal assembly manufactured by the Parker Hannifin Corporation is used between the cask body and the outer closure to prevent the release of radioactive material during the normal conditions of transport and to limit the release of radioactive material during hypothetical accident conditions. This seal assembly is protected by the impact limiter, especially during hypothetical fire accident condition. Because of this protection the temperature of the seal does not exceed the recommended operating temperatures. As stated in Section 2.3.9, the elastomer seals can withstand temperatures up to 700 degrees F for short periods of time. Therefore, during the hypothetical fire accident conditions, containment is maintained. The two penetrations in the bottom of the cask body have an elastomer O-ring and a metal O-ring. These O-ring seals provide the required containment during both the normal conditions of transport and hypothetical accident conditions.

4.2.1.4. Closure

The outer closure is fabricated from high strength alloy steel. Twenty-four (24) socket head cap screws attach the closure to the cask body. A seal test port is located in the closure to allow the containment seal assembly to be tested as required.

4.2.2. Requirement for Normal Conditions of Transport

The radioactive contents of Model FSV-1 in Configurations F or G will be solid, nonfissile, irradiated and contaminated hardware which has been removed from the Fort St. Vrain High Temperature Gas Cooled Reactor (HTGR). Experience to date has shown that such material has a limited amount of loose radioactive contamination that could be available for release from the package during transport.

Information has been obtained from wipes of hardware removed from the reactor regarding the identity and quantity of the radionuclides in the loose contamination. This data is the basis for determining the leakage test requirements to verify the containment requirements in accordance with USNRC Regulatory Guide 7.4 and ANSI N14.5 - 1977.

Identities, quantities, and characteristics of the radionuclides are provided in Table 4-2. The quantity of loose radioactive material per burial canister is based on irradiated hardware with the maximum expected surface area of 60,000 cm^2.

TABLE 4-2

Radionuclide	Millicuries per cm^2	Millicuries per Burial Canister	A_2 Curies	$A_2 \times 10^{-3}$ per Burial Canister
Ag 110m	5.69×10^{-6}	.34	7	.05
Co 57	5.70×10^{-7}	.03	90	.0003
Co 58	4.97×10^{-6}	.30	20	.015
Co 60	4.92×10^{-4}	29.52	7	4.217
Cs 134	6.33×10^{-5}	3.80	10	.38
Cs 137	1.97×10^{-4}	11.82	10	1.182
Fe 59	8.72×10^{-6}	.52	10	.052
Mn 54	5.67×10^{-5}	3.40	20	.17
Zn 65	1.64×10^{-5}	.98	30	.03
	8.45×10^{-4}	50.71		6.096

These radionuclides and their quantities provide a conservative example for representative hardware. Quantities of radionuclides will vary for actual shipments.

The composite A_2 for the mixture of radionuclides is determined as follows:

$$A_2 \text{ composite} = \frac{\text{total curies}}{\text{number of } A_2\text{s}}$$

$$= \frac{50.71 \times 10^{-3}}{6.1 \times 10^{-3}}$$

$$= 8.3 \text{ curies}$$

The regulatory limit for release of radioactive material during the normal conditions of transport is $A^2 \times 10^{-6}$ curies per hour. ANSI N14.5 - 1977 provides the following formula for determining the corresponding allowable leakage rate as follows:

$$L_N = \frac{R_N}{C_N} \quad \text{where,}$$

L_N is the leakage rate in cm^3 per second,

R_N is the release rate in curies per second,

C_N is the specific activity in the package in curies per cm^3,

therefore:

$$R_N = \frac{A_2 \times 10^{-6} \text{ ci}}{\text{hr}} \times \frac{1 \text{ hr}}{60 \text{ min}} \times \frac{1 \text{ min}}{60 \text{ sec}}$$

$$= \frac{8.3 \times 10^{-6}}{3600} = 2.31 \times 10^{-9} \text{ ci/sec}$$

$$C_N = \frac{\text{total curies}}{\text{volume of package (cm}^3\text{)}}$$

$$= \frac{50.71 \times 10^{-3}}{7 \times 10^5} = 7.24 \times 10^{-8} \text{ ci/cm}^3$$

now:

$$L_N = \frac{2.31 \times 10^{-9} \text{ ci/sec}}{7.24 \times 10^{-8} \text{ ci/cm}^3}$$

$$= 3.2 \times 10^{-2} \text{ cm}^3\text{/sec}$$

This is a very conservative containment evaluation since the loose radioactive material is assumed to behave as a gas and no credit is taken for the absence of a driving force.

4.3 CONTAINMENT REQUIREMENTS FOR THE HYPOTHETICAL ACCIDENT CONDITIONS

The regulatory limit for the release of radioactive material following hypothetical accident conditions is A_2 curies in one week. Since the maximum anticipated quantity of radioactive material in a releasable form in Model FSV-1 in Configuration F or G is 50.71×10^{-3} curies and the composite A_2 value for this mixture of radionuclides is 8.3 curies, the entire quantity of radioactive material available for release from the cask could escape without exceeding the regulatory limits.

SECTION 5.0

SHIELDING EVALUATION

5.0 SHIELDING EVALUATION

The shielding evaluation for Model FSV-1 Configuration E is presented in Section 5.1 and the shielding evaluation for Model FSV-1 Configuration F and G is found in Section 5.2.

5.1 Model FSV-1 CONFIGURATION E

5.1.1 Discussion

Shielding requirements for Model FSV-1 in Configuration E are based on the transport of six (6) spent fuel elements from the Fort St. Vrain High Temperature Gas- Cooled Reactor. There is not a significant neutron dose from this type of spent fuel and therefore only gamma shielding is used. This shielding consists of approximately 2 inches of stainless steel and 3.5 inches of depleted uranium.

5.1.2 Results

Radiation dose rates, external to Model FSV-1 in Configuration E are in compliance with the requirements as found in 10CFR71. A tabulation of the calculated radiation dose rates is given in Fig. 5-1.

5.1.3 Source Specification

Source data for each spent fuel element is provided in Table 5-1. The total activity for those isotope of importance to the shielding design and evaluation is approximately 580,000 curies for the six (6) spent fuel elements. This curie content is for spent fuel elements with maximum fission product inventory and 100 days of decay.

Location	Dose Rate (mr/h)
1	Nil
2	1
3	61
4	66
5	268
6	27
7	6
8	7
9	54
10	5.2
11	0.4

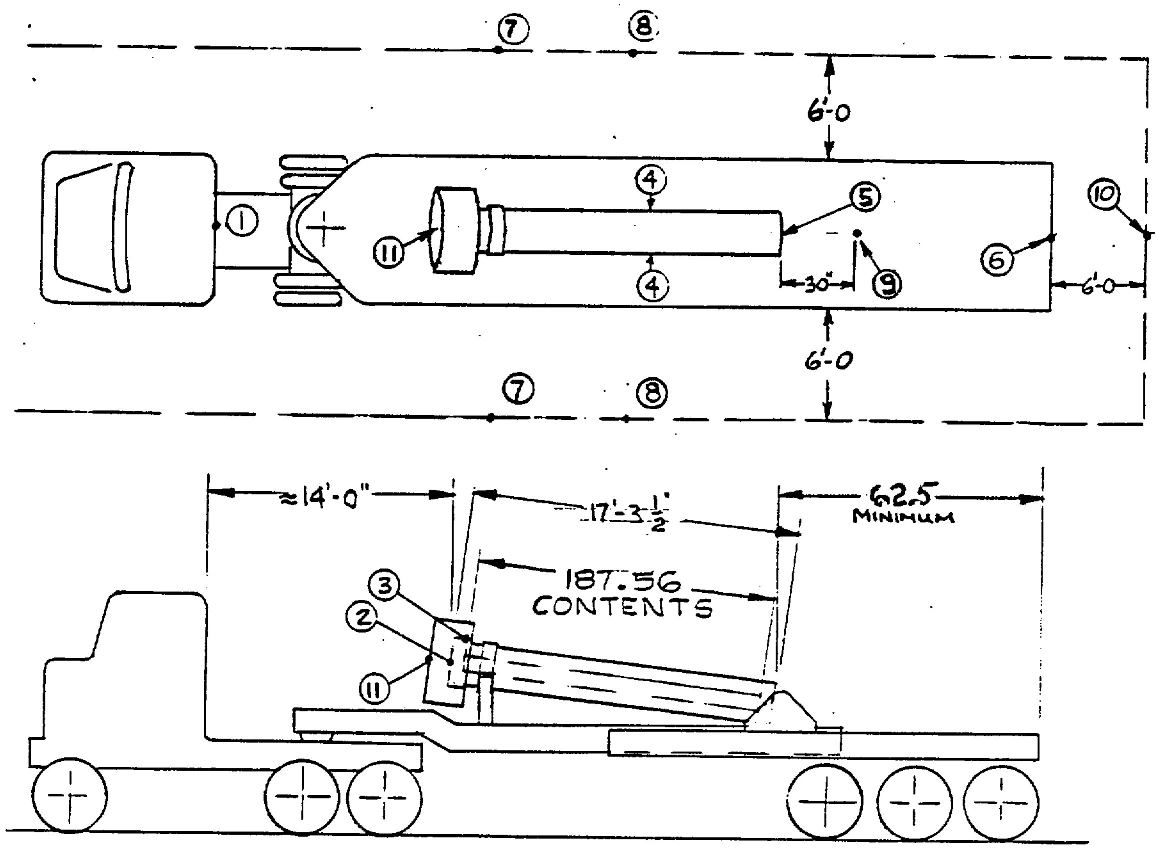


Figure 5-1. Radiation Dose Rates for Model FSV-1 in Configuration E During Transit

TABLE 5-1
SOURCE DATA FOR ONE FUEL ELEMENT

Isotope	Curies*
Y-91	11,900
Zr-95	13,600
Nb-95m	270
Nb-95	13,600
Ru-103	2,700
Ru-106	1,600
Ba-137m	2,080
Ba-140	186
La-140	186
Ce-144	25,200
Pr-144	25,200

*Curie quantities are for fuel element with the maximum inventory fission products after 100 days decay.

5.1.4 Model Specification

The PATH computer code was used to determine the radiation dose rates. Isotopes and associated activities from Table 5-1 were used as input data for the shielding evaluation. All activity was assumed to be uniformly distributed. Dimensions for the package and the thickness of shielding are given in Table 5-2.

5.1.5 Shielding Evaluation

In order to calculate the radiation dose rate for a laminated shield composed of uranium and steel, the dose rate is calculated using buildup factors for steel and then calculated using buildup factors for uranium. The actual dose rate results from adjusting these dose rates by an interpolation technique for buildup factors in a laminated shield.

The following example illustrates the method used for all dose rates computed.

a. Calculation for Dose Rate at Top of Cask

Assume predominant energy of gamma spectrum is 2 MeV (a conservative assumption).

2-7/16 in. Fe = 2.07 mfp (Fe) mfp = mean free path (gamma ray)

2-1/4 in. U = 5.18 mfp (U)

2-7/8 in. Fe = 2.44 mfp (Fe)

TABLE 5-2

DATA FOR MODEL FSV-1 IN CONFIGURATIONS E, F AND G

Inside radius of steel liner	21.11 cm
Thickness of fuel container plus inner liner	2.86 cm
Thickness of uranium at side	8.89 cm
Density of uranium	18.9 gm/cm^3
Thickness of outer shell liner	2.54 cm
Height of fuel column	476. cm
Average density of fuel (assumed to be carbon only)	1.54 gm/cm^3
Steel thickness on cask bottom	27.94 cm
Shield thickness on cask top	
Steel (next to fuel)	3.35 cm
Uranium	10.52 cm
Steel (top of cask)	12.70 cm

Assume 2.07 mfp (Fe) + 5.18 mfp (U) = 7.25 mfp (U)

TOTAL mfp's = 7.25 mfp (U) + 2.44 mfp (Fe) = 9.69 mfp

Br = Buildup

BR-1 for 5 mfp (U) + 2.44 mfp (Fe) [7.44 mfp total] = 5.7

BR-1 for 10 mfp (U) + 2.44 mfp (Fe) [12.44 mfp total] = 9.0

Interpolating

$$\text{BR-1 FOR 9.69 MFP} = 5.7 + \frac{9.59 - 7.44}{12.44 - 7.44}(9.0 - 5.7)$$

BR-1 = 7.2

Dose rate at Top of Cask using PATH and Iron Buildup Factors = 179.5 mR/hr

$$\text{Actual dose rate} = 179.5 \text{ mR/hr} \left(\frac{5.7}{7.2}\right) = 142 \text{ mR/hr}$$

The above technique was used in computing all dose rates.

b. Gamma Buildup in Two-Layer Laminated Shielding

Frequently it is desirable to use two and three layer configurations of iron/concrete; iron/carbon; iron/lead/iron; or iron depleted uranium/iron for gamma shielding. Therefore, sets of curves for these materials at two source energies, 1 MeV and 2 MeV, have been derived. These curves have been used in High Temperature Gas-Cooled Reactor (HTGR) shielding calculations and for computing radiation levels for Model FSV-1 in Configurations E, F and G.

c. Derivation Curves

Dose buildup curves were derived for solid iron, solid lead, and solid uranium first plotted for point isotropic sources of 1 MeV or 2 MeV energy, using data directly from Ref. 5-2. In unpublished work, Chilton has shown that if a log-log grid is utilized with B_r-1 plotted against mean free paths (mfp), the curves will be nearly straight lines. This method has been adopted in the attached figures, and the curves extrapolated to 40 mfp.

Next, the trends displayed in the two-layer iron/concrete configurations in Figs. 21 and 22 of Ref. 5-1 were carefully studied. One set of curves would be applicable to a situation involving a point source like CO^{60} enclosed in a small ID spherical cask. The other set of curves is appropriate to large source geometries and shields with relatively large radii or curvature, up to and including plane sources and slab shields. The latter set of curves is more generally useful for HTGR shield design. Furthermore, the trends established in these curves agree quite well with the results of two other studies (Refs. 5-3 and 5-4).

The curves for two-layer configurations of iron/lead and iron/uranium were derived graphically rather than analytically. Essentially, the envelopes between the curves for solid iron and solid lead, or solid iron and uranium, were subdivided into the two-layer curves using trends and proportions from the iron/concrete curves of Ref. 5-1.

d. Examples:

In practical cases, it will generally be necessary to convert thickness to mfp. The conversions are:

Material	Conversion Factor (mfp/inch) 1 MeV	Conversion Factor (mfp/inch) 2 MeV
Iron	1.19	0.85
Lead	1.97	1.32
Uranium	3.6	2.3

It is important to remember that the curves are plotted for B_r-1 vs. mfp, instead of B_r vs. mfp. Don't forget to add 1.0 to the final buildup result.

Example 1 Shield consisting of 3 inches uranium followed by 2 in. of steel. Source energy = 2 MeV.

$$\left.\begin{array}{l} \text{mfp U} = 3 \times 2.3 = 6.9 \\ \text{mfp Fe} = 2 \times 0.85 = 1.7 \end{array}\right\} \text{total mfp} = 8.6$$

Read from Fig. 5-5

B_r-1 for 5 mfp U followed by 1.7 mfp Fe (i.e., 6.7 mfp total) = 4.6

B_r-1 for 10 mfp U followed by 1.7 mfp Fe (i.e., 11.7 mfp total) = 7.6

Either linear or logarithmic interpolation between 6.7 and 11.7 mfp can be used. Linear interpolation gives:

$$B_r\text{-}1 \text{ for } 8.6 \text{ mfp} = 4.6 + \frac{8.6 - 6.7}{11.7 - 6.7}(7.6 - 4.6) = 5.74$$

Logarithmic interpolation yields:

$$B_r\text{-}1 \text{ for } 8.6 \text{ mfp} = \text{antiln}\left[\ln 4.6 + \frac{\ln \frac{8.6}{6.7}}{\ln \frac{11.7}{6.7}} \ln \frac{7.6}{4.6}\right] = 5.75$$

Therefore, the required building factor is $B_r = 5.75 + 1.0 = \underline{6.75}$

Example 2 Shield consisting of 10 mfp lead followed by 3 mfp steel. Source energy = 1.6 Mev.

Read from Fig. 5-2.

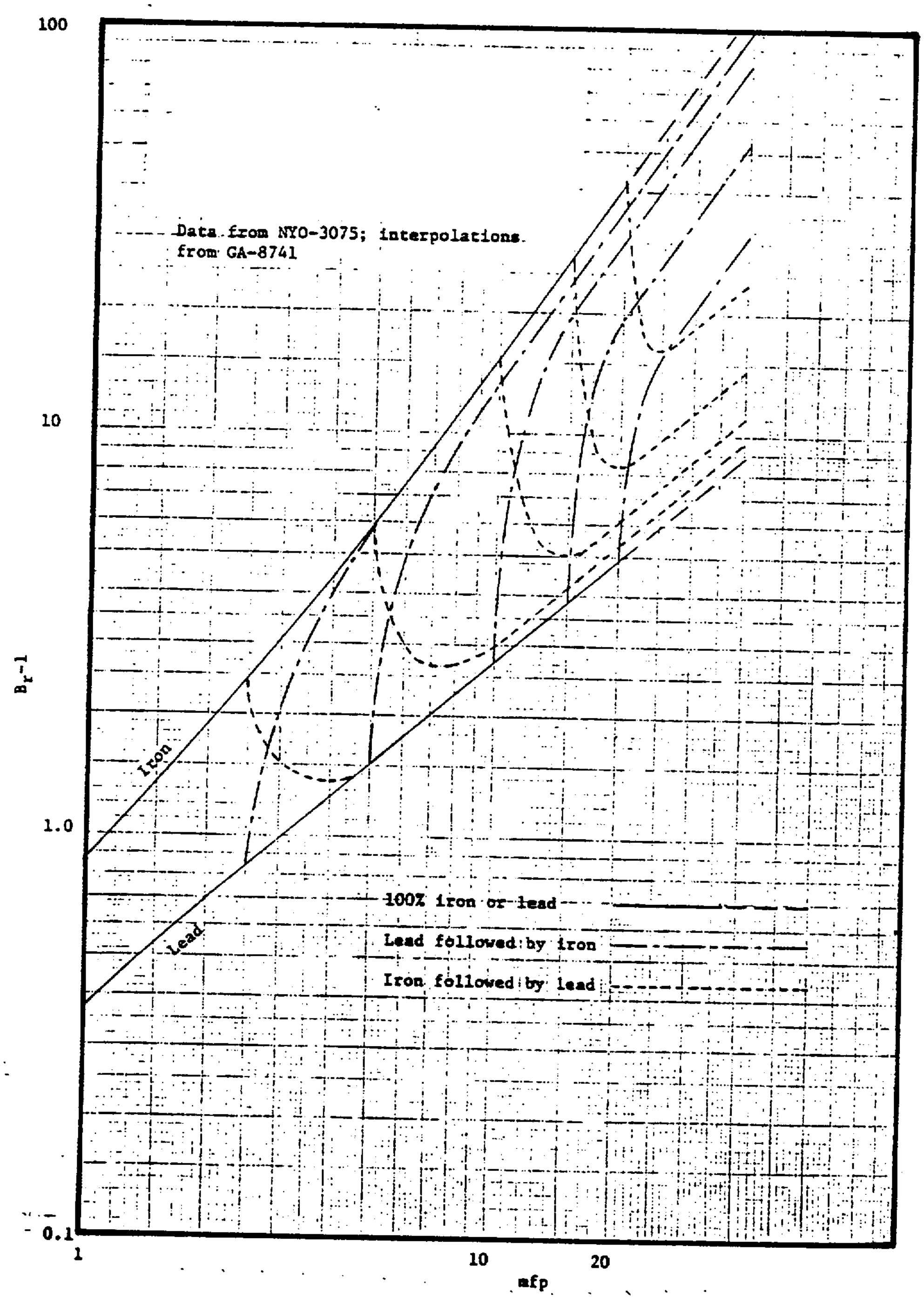


Figure 5-2. Estimated Gamma Dose Buildup Factors in Iron-Lead Configurations (1-MeV Point Isotropic Source)

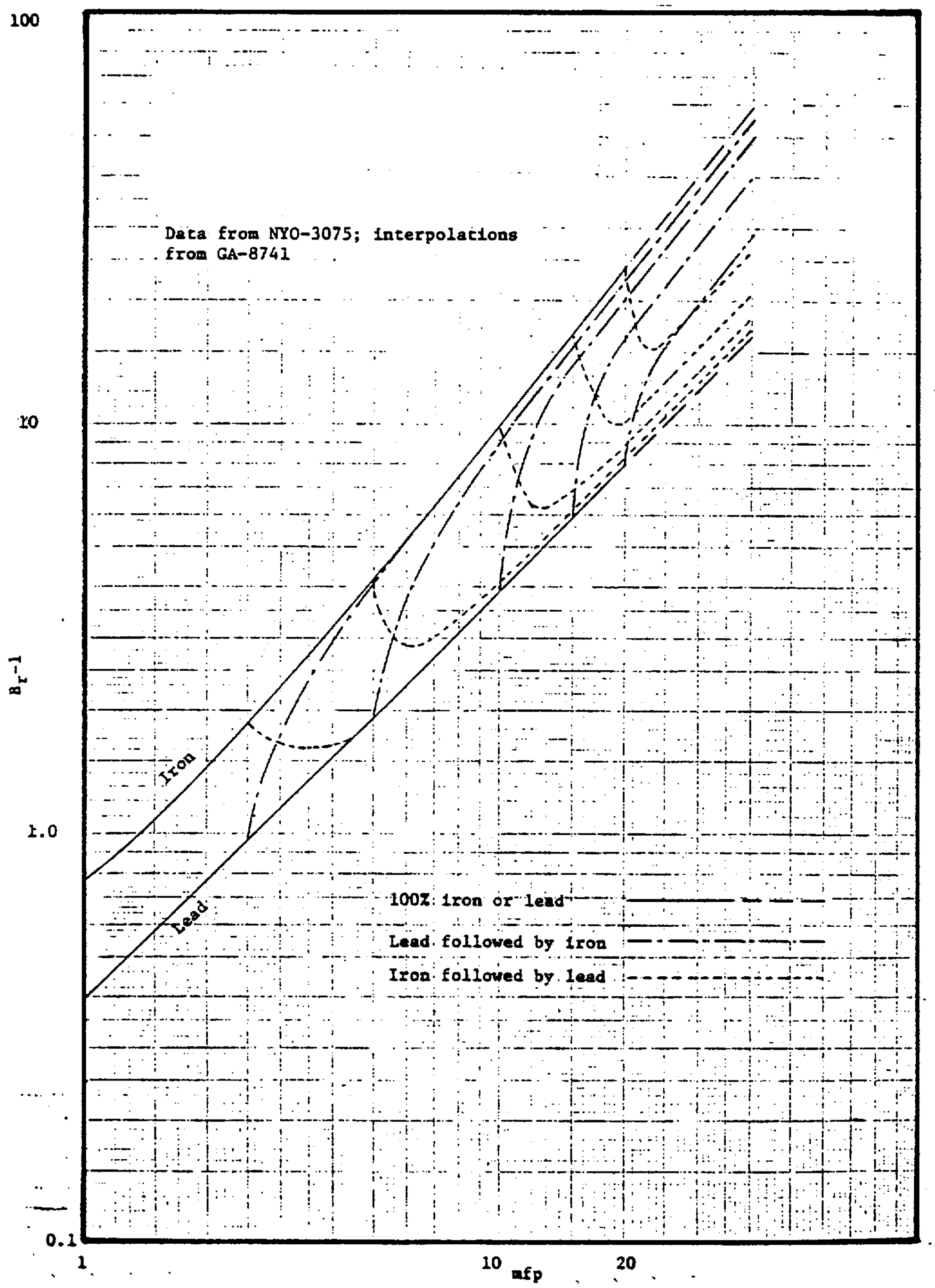


Figure 5-3. Estimated Gamma Dose Buildup Factors in Iron-Lead Configurations (2-MeV Point Isotropic Source)

B_r-1 for 10 mfp Pb followed by 3 mfp Fe, for 1 MeV source energy = 15.0

Read from Fig. 5-3.

B_r-1 for 10 mfp Pb followed by 3 mfp Fe, for 2 MeV source energy = 11.2 using linear interpolation between source energies:

$$B_r\text{-}1 \text{ for } 1.6 \text{ Mev} = 15.0 - \frac{1.6 - 1.0}{2.0 - 1.0}(15.0 - 11.2) = 12.7$$

Logarithimic interpolation gives:

$$B_r\text{-}1 \text{ for } 1.6 \text{ Mev} = \text{antiln}\left[\ln 15.0 - \frac{\ln 1.6}{\ln 2.0} \ln \frac{15.0}{11.2}\right] = 12.3$$

The latter is believed to be more accurate.

Hence, $B_r = 12.3 = 1.0 + \underline{13.3}$

<u>Example 3</u> Shield consisting of 2.5 mfp steel followed by 5 mfp lead followed by 2.5 mfp steel. Source energy = 1.0 Mev.

Three-layer cases like this can be treated, with only small errors, as two-layer configurations, combining the first two materials into a single region. Thus,

2.5 mfp steel + 5 mfp lead --------→ 7.5 mfp lead

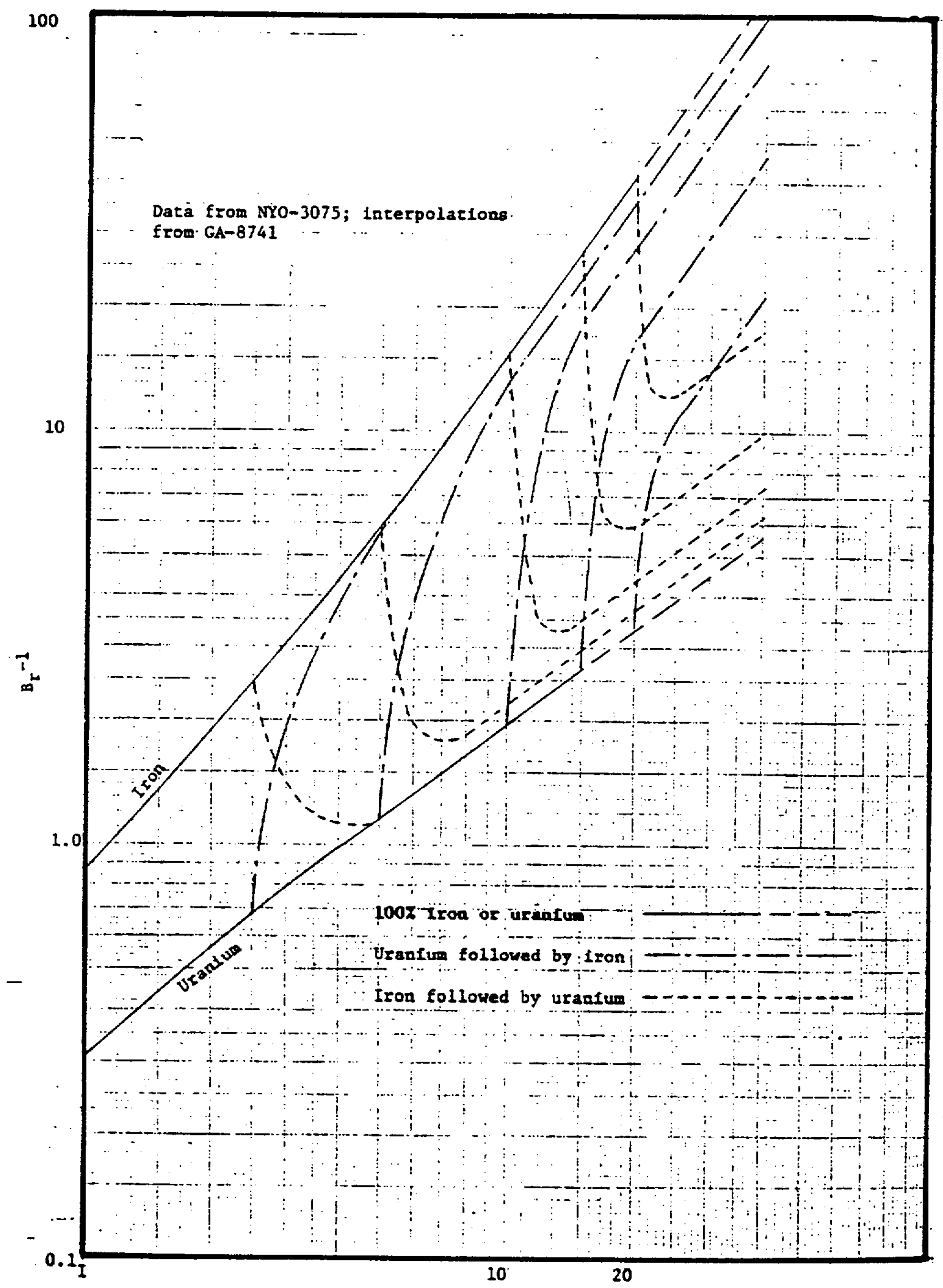


Figure 5-4. Estimated Gamma Dose Buildup Factors in Iron-Uranium Configurations (1-MeV Point Isotropic Source)

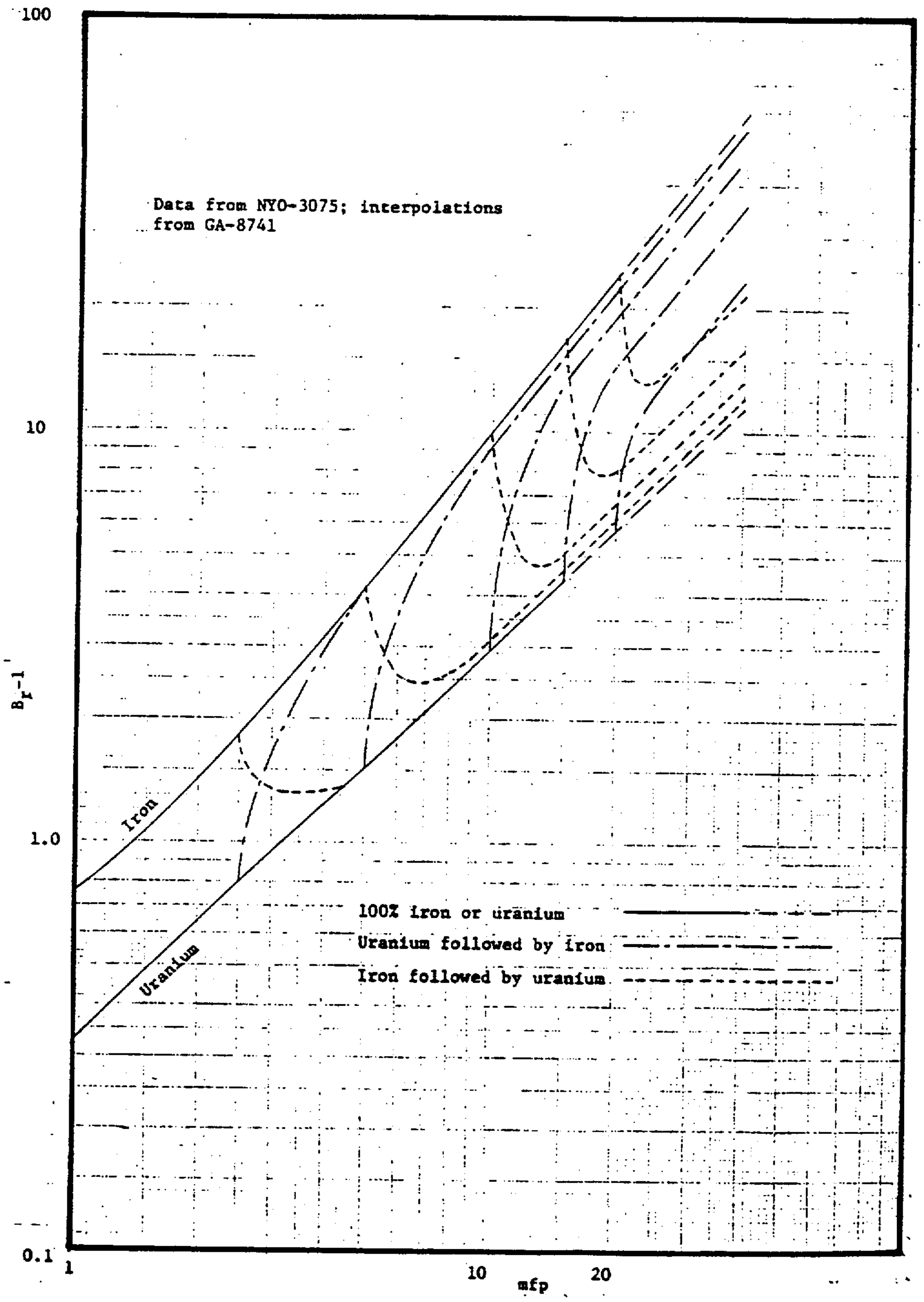


Figure 5-5. Estimated Gamma Dose Buildup Factors in Iron-Uranium Configurations (2-MeV Point Isotropic Source)

Reading Fig. 5-2.

B_r-1 for 5 mfp Pb followed by 2.5 mfp Fe = 8.2

B_r-1 for 10 mfp Pb followed by 2.5 mfp Fe = 13.3

Interpolation gives:

B_r-1 for 7.5 mfp Pb followed by 2.5 mfp Fe = 10.4

Therefore,

$B_r = 10.4 + 1.0 = \underline{11.4}$

5.2 Model FSV-1 Configurations F and G

5.2.1 Discussion

Model FSV-1 in Configurations F and G are designed for the transport of solid, nonfissile, irradiated and contaminated hardware. The exact description of every possible mixture of irradiated hardware which may be transported is not known at this time, therefore a shielding evaluation for all potential contents has not been prepared. However, a detailed shielding evaluation has been completed for identified contents which are expected to have some of the higher dose rates.

The source term of 11,000 curies of Co 60 plus Co 58 does not constitute a curie limit for Model FSV-1 in Configuration F and G. The curie limit for the package is controlled by the regulatory limits for external dose rate.

The top closure plug of the burial canister provides supplemental shielding for Model FSV-1 Configurations F and G. Both these pieces are necessary to satisfy the external dose rate limits for the normal conditions of transport and also must remain effective following the hypothetical accident conditions to satisfy the regulatory limits.

A spacer that provides additional supplemental shielding (see Drawing GADR 55-2-13) is used within the burial canister for Configuration G. This additional supplemental shielding is necessary to satisfy the regulatory limits during the normal conditions of transport but is not necessary following the hypothetical accident conditions.

5.2.2 Results

The calculated radiation dose rates, based on 11,000 curies, are shown below for the various locations. Radiation dose rates, external to Model FSV-1 in Configurations F and G are in compliance with the requirements as found in 10CFR71.

Location	Radiation Dose Rate
Cask surface	138 mR/hr
Two meters from edge of trailer	5.6 mR/hr
Drivers location in tractor	less than 1 mR/hr
Surface of burial canister	19000 R/hr

5.2.3 Source Terms

The radionuclides, Co 60 plus Co 58 contribute all of the significant gamma radiation from the irradiated hardware in Model FSV-1 Configurations F and G. For the shielding calculations a total source, for these two radionuclides, of 11,000 curies was used.

5.2.4 Shielding Evaluation

The shielding calculations were performed using the PATH computer program which is described in Section 5.3. The radioactive contents consist mainly of control rod cans and miscellaneous hardware fabricated from Inconel-600, Inconel-800, and various steels. The computer model was configured to describe a distributed source rather than a line source or a point source.

REFERENCES

5-1 Engholm, B. A., "Gamma Buildup in Heterogeneous Media," Gulf General Atomic Incorporated Report GA-8741, Sept. 16, 1968.

5-2 Goldstein, H., and J. W. Wilkins, Jr., "Calculations of the Penetration of Gamma Rays," NYO-3075, June 30, 1954.

5-3 Miyasaka, S., and A. Tsuruo, "Dose Buildup Factors of Multilayer Slabs for a Point Isotropic Source," J. Nucl. Sci. Tech. 3, 9, 393, Sept. 1966.

5-4 Futtermenger, W., et al., "To the Calculation of Gamma Ray Buildup Factors in Multilayered Shield," AERE-R-5773, Vol. 2, 1968.

5.3 APPENDIX

The PATH code, primarily a gamma shielding computer program, utilizes the common point-kernel integration technique to perform calculation of dose rates and shielding requirements for complex geometry and various source types. The code has been in production use for in-house shielding analysis and design work at GA Technologies for more than ten years.

The heart of the PATH code is the geometry routine, which defines the source-shield configuration by a set of possibly overlapping regions of simple shapes with a mother-daughter ordering scheme, and determines the path length in each region by a direct method. Regions are available in various shapes with any axial orientation including prisms, cylinders, spheres, and frustra of cones. Shield regions may be redefined between dose point calculations for making parameter studies.

The options of source types consist of point, line, disc, polygon, shell, cylinder, and prism sources in any spatial orientation. In addition, the X-Y-Z meshing mode is applicable for modeling geometrically complex source regions such as sphere, hemisphere, quarter cylinder, etc. The source terms can be described in two ways: source strengths (MeV/sec or photons/sec per unit mesh) at given energy levels, and/or isotopes with associated activities (curies per unit mesh). The latter option can lead to output of the percent contribution to the total dose rate from each isotope for identifying the important contributors.

SECTION 6.0

CRITICALITY EVALUATION

6.0 CRITICALITY EVALUATION

Model FSV-1 Configurations F and G are not evaluated for the transport of fissile material.

6.1 Discussion and Results

6.1.1 Discussion

Model FSV-1 Configuration E is designed for the transport of six (6) spent fuel elements from the Fort St. Vrain, High Temperature Gas-Cooled Reactor.

The fissile material is contained in fuel elements which are hexagonal in cross section with dimensions 14.2 in. across flats by 31.2 in. high as shown in Fig. 6-1. Each fuel element contains coolant and fuel channels which are drilled from the top face of the element. Fuel holes are drilled to within about 0.3 in. of the bottom face and are closed at the top by a 0.5 in. cemented graphite plug. The fuel channels occupy alternating positions in a triangular array within the element structure, are 0.5 in. in diameter and contain the active fuel.

The element structure consists of needle coke and/or isotropic graphite. The fuel itself is in the form of carbide particles coated with layers of pyrolytic carbon and silicon carbide. The fuel bed contains a homogeneous mixture of two types of particles, called fissile and fertile. Fresh fissile particles contain both thorium and 93.5% enriched uranium, while fresh fertile particles contain only thorium. The important parameters of fresh particles are:

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 6-1. HTGR Standard Fuel Element

Parameter	Fissile	Fertile
Nominal Th/U Ratio	3.6 or 4.25	All Th
Particle Composition	$(Th/U)C_2$	$Th\ C_2$
Average Fuel Particle Diam, μm	200	450
Average Total Coating Thickness, μm	130	140

Irradiated fuel elements contain, besides fission products, thorium, U-233, U-235, other uranium isotopes and a small quantity of plutonium. In the fertile particles, the fissile material is essentially U-233, while the fissile particles contain the residual U-235 and bred U-233.

The effective fissile material enrichment (U-235/U+Th) in fresh fuel for the initial core and reload segments varies between 2% and 12% due to radial and axial fuel zoning requirements. The most reactive fresh fuel element contains a maximum of 1.4 Kg of 93.5% enriched uranium and about 11.3 Kg of thorium. Any irradiated elements will contain a smaller amount of fissile material, since the conversion ratio of the reactor is less than unity.

6.1.2 Results

During the normal conditions of transport, the multiplication constant or K_{eff} is 0.41.

During the hypothetical accident conditions with non-flooded spent fuel elements, the K_{eff} is 0.41 and for the flooded spent fuel particle case the K_{eff} is 0.89.

6.2 Package Fuel Loading

6.2.1 Assumptions

The following assumptions were used for the criticality evaluation:

a. The fuel elements contain the most reactive fresh fuel composition anticipated for fuel shipment, i.e., a maximum of 1.4 Kg of 93.5% enriched uranium and about 11.3 Kg of thorium per element resulting in a maximum of 8.4 Kg of uranium and about 68 Kg of thorium per package. The total amounts of all materials present in a fuel element as well as the homogenized atom densities (over a fuel element) are summarized in Table 6-1.

b. The presence of burnable poison or other neutron absorbing material, other than U-235, U-238, thorium, silicon and graphite, is neglected.

c. The fuel is at room temperature.

d. All fission products are neglected.

These assumptions are all conservative. In general spent fuel elements will contain considerably less fissile material, Also, all other fresh fuel element types are less reactive due to their lower uranium contents and/or higher thorium content. Hence, elements with a maximum of 1.4 Kg of uranium and a thorium/uranium ratio of at least 8.1/1 are acceptable.

TABLE 6-1
MOST REACTIVE FUEL ELEMENT MATERIAL CONTENTS

	Total Amount (Kg)	Homogeneized Atom Density (atoms/b-cm)
Th-232	11.25	3.28×10^{-4}
U-235	1.31	3.76×10^{-5}
U-238	0.078	2.22×10^{-6}
Silicon	4.63	1.12×10^{-3}
Carbon	111.3	6.27×10^{-2}

6.3 Model Specification

a. Geometry

Two geometric models were used to evaluate the criticality situation for the shipping cask. The one-dimensional model is shown in Fig. 6-2, and assumes an infinitely long cylinder. This model is adequate as long as it can be assumed that the fuel is well contained within the fuel elements.

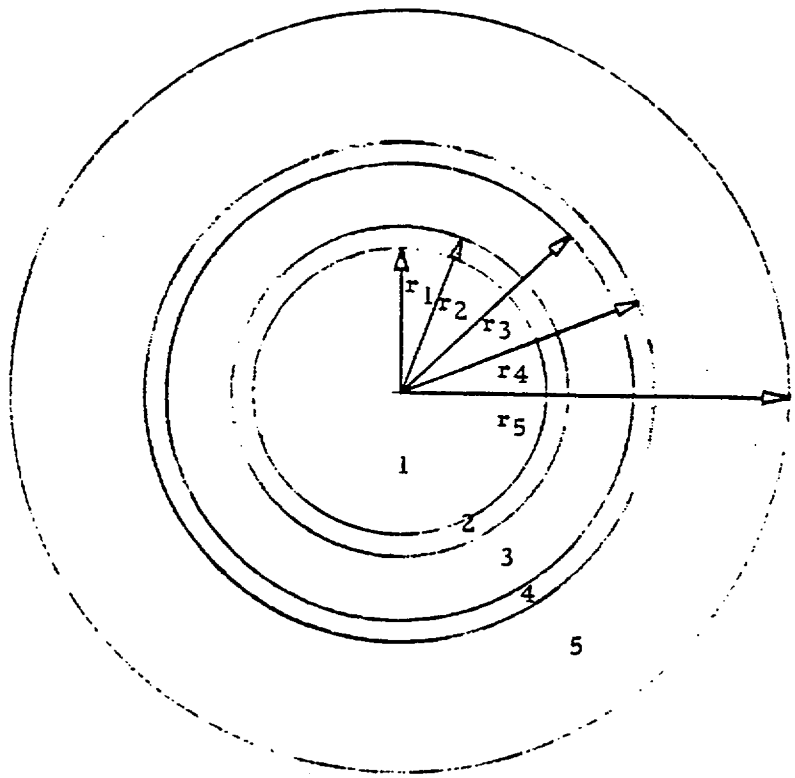
The two-dimensional geometric model, shown in Fig. 6-3, was used for the maximum criticality situation which includes fuel element breakage, internal flooding and an accumulation of fuel particles at the bottom of the cask.

b. Cross Sections

Cross sections were calculated with the GGC-4 code, a description of which is provided in the appendix. The calculational methods and the basic nuclear cross section data are well established and in use for the high temperature gas-cooled reactor (HTGR) nuclear design.

c. Computer Codes

The DTF-IV transport code was used for one-dimensional and the GAMBLE-5 code was used for the two-dimensioned calculations. Abstracts of both codes are provided in the Appendix.



Region	Region Outer Radius	Region Contents
1	r_1 = 21.11 cm	Fuel element (+ water)
2	r_2 = 23.97 cm	Iron
3	r_3 = 32.86 cm	Uranium-235 & uranium-238
4	r_4 = 35.40 cm	Iron
5	r_5 = 55.40 cm	Void or water

Figure 6-2. Model FSV-1 Configuration E - One Dimensional Model

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 6-3. Model FSV-1 Configuration E - Two Dimensional Model

6.4 Criticality Calculations

6.4.1 Normal Conditions of Transport

During normal transport not more than 6 fuel elements will be stacked end to end within the inner container. No hydrogeneous or other moderating material besides the structural and coating graphite will be present.

The fuel elements occupy 81.4% of the inner container of the shipping cask. The contents of the elements were homogenized over the inner container of the cask and used in one-dimensional transport calculations. Material concentrations for the different regions shown on Fig. 6-2 are given in Table 6-2.

The calculated multiplication constant (k_{eff}) for this condition, assuming an infinitely long container, is 0.40 and no criticality hazard is to be expected under any circumstances.

6.4.2 Hypothetical Accident Conditions

a. Unflooded Fuel

For the hypothetical accident conditions some breakage of the fuel elements has to be expected with possible accumulation fuel particles in a corner of the cask. Detailed drop test data for irradiated fuel elements, however, are not available and some assumptions have to be made concerning the amount of particles released from the fuel elements. Assuming an upper limit of 20% particle loss and accumulation of these particles in coolant holes and the space between container wall and fuel element at the

TABLE 6-2
MATERIAL CONCENTRATIONS

Region	Material	Concentration (atoms / b-cm)
1	Th-232	2.67×10^{-4}
	U-235	3.06×10^{-5}
	U-238	1.81×10^{-6}
	Si	9.12×10^{-4}
	C	5.10×10^{-2}
2	Fe	8.50×10^{-2}
3	U-238	4.8×10^{-2}
	U-235	9.6×10^{-5}
4	Fe	8.5×10^{-2}
5	Void	--

bottom of the container, the multiplication constant is calculated to be less than 0.41, with the cask completely immersed in water.

In the analysis it was assumed that the fuel particles fall out of the fuel channels into the coolant channels and spaces between element and container wall and form a homogeneous mixture with the remaining fuel element graphite. At 10% particle loss the lowest 24 cm of the cask would be filled with fuel particles and fuel elements. The remaining elements would have a correspondingly reduced fuel loading. At 20% particles loss, the lowest 48 cm would be filled with particles. Material concentrations for the different fuel regions are shown in Table 6-3 consistent with the geometrical model of Fig. 6-3.

The following multiplication constants were calculated with the two-dimensional model:

No particle loss	$k_{eff} = 0.37$
10% particle loss	$k_{eff} = 0.37$
20% particle loss	$k_{eff} = 0.41$

The first result agrees well with the data from the one-dimensional transport calculation. The other results show that partial particle loss from the fuel elements and subsequent particle accumulation at the bottom of the cask do not significantly increase the multiplication constant. Even if the whole cask were filled with particles only, the overall multiplication constant of the immersed cask would be less than 0.55. The presence of water at the outside of the cask

TABLE 6-3
CONCENTRATIONS FOR MOST REACTIVE CONDITION (UNFLOODED)

Region	Material	Concentration (atoms/b-cm) 10% Particle Loss	20% Particle Loss
1	Th-232	2.4×10^{-4}	2.14×10^{-4}
	U-235	2.75×10^{-5}	2.45×10^{-5}
	U-238	1.63×10^{-6}	1.45×10^{-6}
	Si	8.21×10^{-4}	7.30×10^{-4}
	C	5.01×10^{-2}	4.91×10^{-2}
2	Th-232	7.41×10^{-4}	
	U-235	8.49×10^{-5}	
	U-238	5.02×10^{-6}	
	Si	2.53×10^{-3}	
	C	6.92×10^{-2}	
3	Fe	0.085	
4	U-238	0.0478	
	U-235	0.000096	
5	H	0.0668	
	O	0.0334	

Height of Particle Accumulation: 10% loss 24 cm
20% loss 48 cm

has no significant effect on the criticality of the system. The iron and uranium shield acts as a sink for thermal neutrons. Fast neutrons escaping from the fuel region are moderated in the water and absorbed before they can return to the fuel.

b. Flooded Fuel

In order to obtain an upper limit for the multiplication constant of the shipping cask under flooded conditions the following assumptions were made in addition to those stated in Section 6.2.1:

(1) All fuel particles leave the fuel elements (100% particle loss).

(2) The fuel particles accumulate in the fuel holes, coolant holes and void spaces between fuel elements and container wall.

(3) The graphite structure of the fuel elements stays intact allowing the highest concentration of fissile material in the available void space.

(4) All void space between the fuel particles and in the unfueled section of the cask is filled with water.

In particular, the following cases were considered:

Case 1: All fuel particles accumulate at the bottom of the cask, filling the available void space. The particle packing fraction is 0.65. The overall height of the fueled section is 171 cm.

Cases 2, 3, 4: The fuel particles float in water. A homogeneous mixture of water and particles occupies the available void space up to a height of 214 cm, 257 cm and 343 cm, respectively.

Case 5: The fuel particles float in water. A homogenous mixture of water and particles fills all available void space in the shipping cask.

Material concentrations for these 5 cases consistent with Fig. 6-3 are given in Table 6-4. The calculated multiplication constants for these cases are as follows:

Fuel Containing Section

Case	Height (cm)	H/U-235	k_{eff}
1	171	143	0.84
2	214	244	0.89
3	257	346	0.89
4	343	551	0.85
5	476	866	0.77

Figure 6-4 shows these results graphically. The most reactive situation, a multiplication constant of about 0.9, is obtained if the total fuel contents, in a mixture of water and particles, occupies the available void spaces of about half the cask (235 cm). From these

TABLE 6-4
CONCENTRATIONS FOR MOST REACTIVE FLOODED CONDITIONS
(Calculational Model of Fig. 6-5)

Region	Material	Case I	Case 2	Case 3	Case 4	Case 5
		Concentrations (atoms/b-cm)				
1	C	4.09-2	4.09-2	4.09-2	4.09-2	--
	H	3.47-2	3.47-2	3.47-2	3.47-2	--
	O	1.74-2	1.74-2	1.74-2	1.74-2	--
2	Th-232	7.41-4	5.93-4	4.94-4	3.71-4	2.67-4
	U-235	8.49-5	6.79-5	5.66-5	4.25-5	3.06-5
	U-238	5.02-6	4.02-6	3.35-6	2.51-6	1.81-6
	Si	2.53-3	2.03-3	1.69-3	1.27-3	9.12-4
	C	6.92-2	6.35-2	5.98-2	5.50-2	5.11-2
	H	1.21-2	1.66-2	1.96-2	2.34-2	2.65-2
	O	6.05-3	8.30-3	9.80-3	1.17-2	1.33-2

Other regions as in Table 6-3

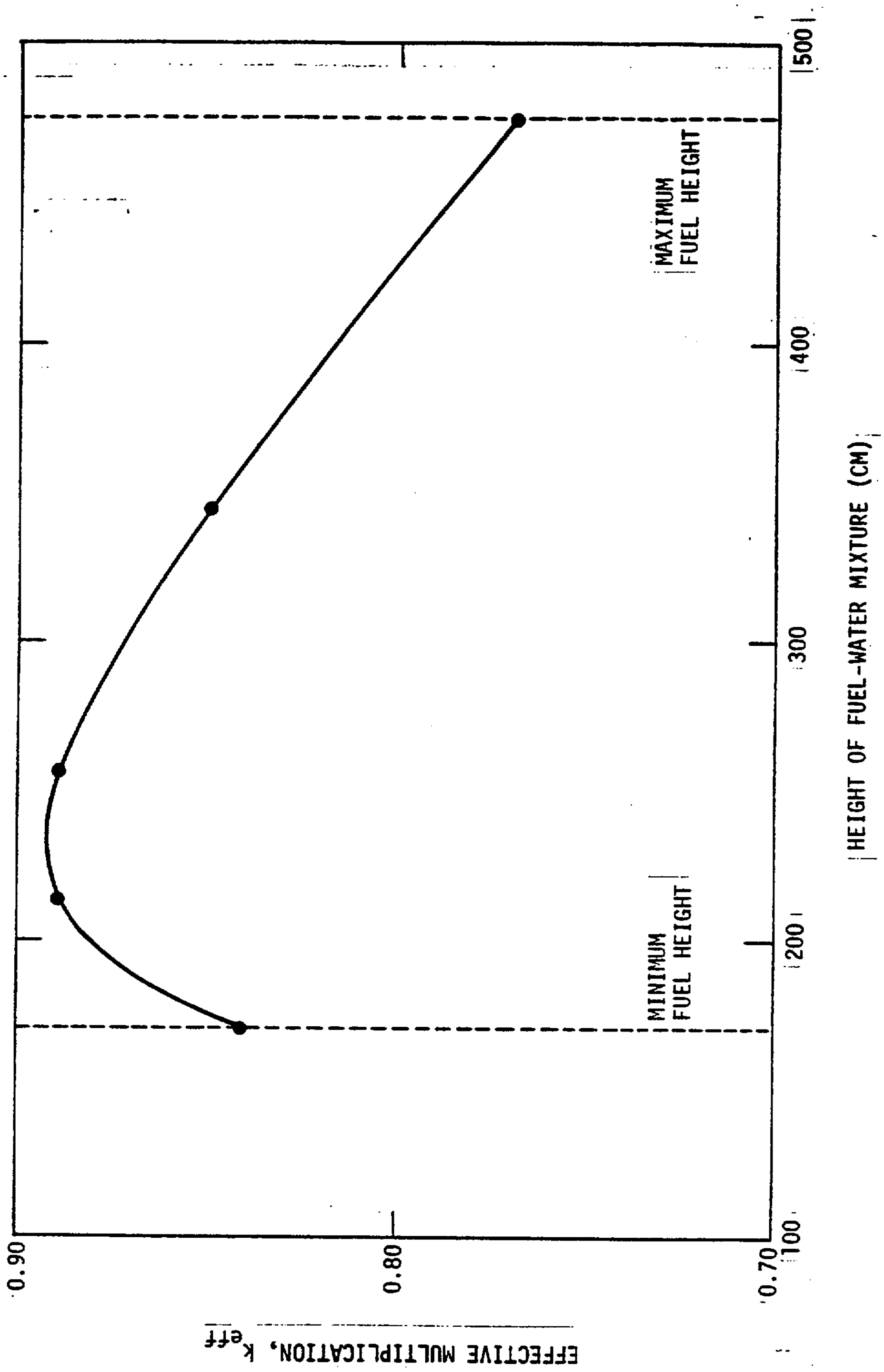


Figure 6-4. Model FSV-1 Configuration E with Internal Flooding
Contents: 8.4 kg 93.5% Enriched Uranium
68.0 kg Thorium-232

results it is concluded that no critical arrangement can occur even if all the fuel leaves the fuel elements and the cask is completely flooded.

6.5 Criticality Benchmark

The calculational methods used for the criticality analysis are essentially the same as those used for HTGR design and the analysis of the HTGR critical facility. Comparisons of experimental and calculational results are well documented in the Final Hazards Report for the Fort St. Vrain Reactor. These results indicate that the accuracy of the methods used is in the order ± 0.02 Δk.

6.6 Appendix

6.6.1 Designation of Program - GGC4.

a. Computer for which program is designed and others upon which it is operable - Univac 1108.

b. Nature of physical problem solved - the GGC4 program solves the multigroup spectrum equations with spatial dependence represented by a single positive input buckling. Broad group cross sections (shielded or unshielded) are prepared for diffusion and transport codes by averaging with the calculated spectra over input-designated energy limits. The code is divided into three main parts. A fast (GAM) section which covers the energy range from 14.9 MeV to 0.414 eV, a thermal (gather) section which covers the energy range from 0.001 to 2.38 eV, and a combining (combo) section which combines fast and thermal cross sections into single sets. Basic nuclear data for fast section which consists of fine group-averaged cross sections and resonance parameters is read from a data tape. The fine group absorption and fission cross sections may be adjusted by performing a resonance integral calculation. Utilizing a fission source and an input buckling, the code solves the P1, B1, B2, or B3 approximation to obtain the energy-dependent fast spectrum. Two or six spatial moments of the spectrum (due to a plane source) may also be evaluated. Instead of performing a spectrum calculation, the user may enter the legendre components of the angular flux directly. For as many input-designated broad group structures as desired, the code calculates and saves (for the combining section) spectrum weighted averages of microscopic and macrosocpic cross sections and transfer arrays. Slowing down sources are calculated and saved for use in the lower energy range. Given basic nuclear data, the thermal section of GGC4 determines a thermal spectrum by either reading it as input, by calculating a Maxwellian spectrum for a given temperature, or by an iterative solution of the P0, B0, P1, or B1 equations for an input buckling. Time moments of the time and energy-dependent diffusion equations are calculated (as an option) using the input buckling to represent leakage. Broad group cross sections are prepared by averaging fine group cross sections over the calculated spectra. Broad group structures are read as input. The combining section of GGC4 takes the broad group-averaged cross sections of GGC4 takes the broad group-averaged cross sections from the fast and thermal portions of GGC4 and forms multigroup cross section tables. These tables are prepared in standard formats for transport or diffusion theory calculations. In addition, it is possible to use the combining section to produce mixtures not used in the spectrum calculation or to combine the results of different fast and thermal section calculations and so on. These options are described in Reference 2.

c. Method of Solution - In the fast section either the P1 or the B1, B2, or B3 approximation is made to the transport equation using the positive, energy-independent buckling. In each approximation legendre moments of the angular flux are computed by direct numerical integration of the slowing down equations. In the resonance calculations, doppler broadened (at an input temperature) absorption and scattering cross sections are used. The resonance treatment allows up to two admixed moderators in an absorber lump imbedded in a surrounding moderator. The absorber in the lump is treated by using either the narrow resonance approximation, the narrow resonance infinite mass approximation, or a solution of the slowing down integral equations to determine the collision density using either an asymptotic form of, or an integral equation solution for, the collision density. In the resonance calculation either standard geometry collision probabilities are used or tables of collision probabilities are entered. Dancoff corrections can also be made. In the region of unresolved resonances, resonance absorption is calculated by using Porter-Thomas distributions, but only S-wave neutrons are considered. In the thermal section either the B0, B1, P0, or P1 approximation to the transport equation is made, and in all options legendre moments of the angular flux are computed. A trapezoidal energy integration mesh is used, and the resulting equations are solved iteratively by using a source-normalized, over-relaxed, Gaussian technique. Averages over broad groups are performed by simple numerical integration. The results obtained in the fast and thermal sections are stored on special tapes.

These tapes may contain results for a number of problems, each problem including fine group cross section data for a number of nuclides. If the problem number is specified on these tapes, and a desired list of nuclides is given, the combining code will punch microscopic cross sections for the requested list of nuclides. The program also treats mixtures. Given the atomic densities of the nuclides in a mixture, the code will punch macroscopic cross sections.

d. Restrictions on the Complexity of the Problem - Maximum of -

99 fast groups
101 thermal fine groups
99 fast broad groups
50 thermal broad groups
50 broad groups in the combining section
250 resonances per nuclide
2 moderators admixed with a resonance absorber
305 entries in the escape probability table for cylindrical geometries
505 entries in the escape probability table for slab geometries a single and positive value for the buckling (B2) must be supplied.

e. Typical Running Time - A 81 calculation in the fast section for 3 nuclides and 6 broad groups takes approximately 4 minutes on the Univac-1108 if a resonance calculation (1/2 minute) is performed for one nuclide. The thermal calculation for 3 broad groups requires approximately 2 minutes, which includes about 7 seconds for the iterative procedure. To punch standard diffusion and standard transport cross sections for this problem requires 2 sec.

f. Unusual features of the program - there is an option in GGC4 which makes it possible to shorten the punching process for large two-dimensional transfer arrays. This can be done by specifying a maximum number of desired upscattering and downscattering terms.

g. Related and auxiliary programs - GGC4 is a revision of the earlier program, GGC3. To prepare, handle, and update the basic cross section tapes which are used as input for GGC4, the following codes are utilized--HAKE, MST, PRINT, MIXER, WTFG, MGT3, SPRINT, COMBIN, and DOP.

h. Status - Production

i. References -

J. Adir and K. D. Lathrop, Theory of Methods Used in GGC-3 multigroup cross section code, GA-7156, July 1967.

J. Adir, S. S. Clark, R. Froehlich, and L. J. Todt, Users and Programmers Manual for GGC-3 Multigroup Cross Section Code, Parts 1 and 2, GA-7157, July 1967.

M. M. Drake, Description of Auxiliary Codes Used in the Preparation of Data for the GGC-3 Code, GA-7158, August 1967.

J. Adir, GGC4 Input Instructions, GA Memorandum, June 20, 1968.

BCDCON, GA Note.

j. Machine Requirements - 64K memory with 11 tape units (some of which may be drum areas).

k. Programming language used - Fortran IV.

l. Operating system or monitor under which program is executed - Univac Exec. II, GAX29.

m. Any other programming or operating information or restrictions - there is no restriction on the number of problems that can be run consecutively in each section, nor is there a restriction on the number of nuclides per problem. Without using the new options, GGC4 can also be run with the GGC3 input instructions.

n. Author of the Abstract - J. Adir.

6.6.2 Designation of Program - 1DF

(DTF- IV is essentially the same.)

a. Computer for which code is designed - Univac 1108, IBM-7030, CDC-6600, IBM-7094.

b. Nature of physical problem solved - the linear time-independent Boltzmann equation for particle transport is solved for the energy, space and angular dependence of the particle distribution in 1-D slabs, cylinders, and spheres. Independent source or eigenvalue (multiplication, time absorption, element concentration, zone thickness, or system dimension) problems are solved subject to vacuum, reflective, periodic, or white boundary conditions. A complete energy transfer scattering matrix is allowed for each legendre component of scattering. Solutions to the adjoint transport equation are also obtained.

c. Method of Solution - Energy dependence is treated by the multi-group approximation and angular dependence by a general discrete ordinates approximation. Anisotropic scattering is approximated by a truncated spherical harmonics expansion of the scattering kernel. Within-group scattering and upscattering iteration processes are accelerated by system-wide renormalization procedures. Chebyshev acceleration is automatically applied to accelerate inner iteration convergence. At the option of the user, Chebyshev acceleration factors can be entered as input. Approximations and iterative cycles have been described in detail by Lathrop (Ref. 1 below). 1DF and DTF-IV are essentially the same.

d. Restrictions of the complexity of the problem - the variable dimensioning capability of Fortran IX has been utilized so that any combination of number of groups, number of spatial intervals, size of angular quadrature, etc., can be used that will fit within the total core storage available to a user.

e. Typical running time - a few minutes.

f. Unusual features of the program - anistropic distributed sources may be used and the incoming angular flux at the right boundary may be specified.

g. Related and auxiliary program - GTF, 2DF, and TWOTRAN.

h. Status - Production.

i. References

Lathrop, K. D., 'DTF-IV, a Fortran-IV Program for Solving the Multigroup Transport Equation with Anistropic Scattering, 'USAEC Report LA-3373, Los Alamos Scientific Laboratory, 1965.

j. Machine requirements - no special requirements and no auxiliary storage required.

k. Programming language - Fortran IV.

l. Operation system - Univac Exec II.

m. Other programming information - all storage requirements are computed in the main program. By performing the additions indicated by the algorithm, the precise amount of storage required for a problem can be determined.

n. Author of the abstract - taken from the book 'Nuclear Design Methods in Use at General Atomic.' Modified slightly 9/1/69.

6.6.3 Designation of Program - GAMBLE-5

a. GAMBLE-5, is a program for the solution of the multigroup neutron diffusion equations in two dimensions with arbitrary group scattering.

b. Computer for which program is designed - Univac 1108.

c. Nature of physical problem solved - the homogenous two-dimensional multigroup diffusion theory equations with arbitrary group-to-group scattering and arbitrary fission transfer are solved for heterogenous assemblies in X-Y and R-Z geometry. Homogenous logarithmic boundary conditions are used at the outer surface of the assembly and at the surface of nondiffusion regions. The results include the group- and point-dependent neutron fluxes, the power distribution, the neutron multiplication factor (k-eff), and a detailed neutron balance.

d. Method of solution - The multigroup diffusion theory equations are approximated by five-point difference equations for an arbitrary nonuniform mesh grid. The system of difference equations is solved by an extension of the power method to find the eigenvector (neutron flux) and the eigenvalue (k-eff). Successive line overrelaxation is applied in a special form (exponential overrelaxation) that guarantees the nonnegativity of the neutron flux. Coarse mesh rebalancing is used to improve the preasymptotic convergence behavior. A variation of Aitkens' method is used to improve the asymptotic convergence behavior, assuming only one error mode.

e. Restrictions on the complexity of the problem -
Maximum number of energy group - 10
Maximum number of space meshpoints - 20,000
Maximum number of different material regions - 255

f. Typical running time - A seven-group problem (three fast groups and four thermal groups) in (R,Z) geometry with 2842 space mesh points took 82 iterations assuming a tight convergence criteria (maximum relative flux change less than 0.000007). The total running time (including extensive output) on the Univac 1108 was 12 min.

g. Unusual features of the program -

(1) The coarse mesh rebalancing scheme makes possible the successful solution of difficult problems for which certain group-mesh points are both strongly and weakly coupled to some of their neighbors (e.g., highly nonuniform mesh spacings or material properties, air gaps, cell problems with weak group coupling, etc.).

(2) Simultaneous performance of computation and data transfer with virtually no delay caused by the use of drum storage.

(3) Ability to do efficient restarts for longer running problems and the ability to accept a flux guess on tape from a similar problem.

h. Related and auxiliary programs - GAMBLE-5 is a major revision of the GAMBLE-4 code. Some of the essentials of the iterative technique used have been adopted from exterminator.

i. Status - The program has been in production use since August 1967 and may be obtained by domestic users from the Argonne code center.

j. References -

(1) J. P. Dorsey and R. Forehlich, 'GAMBLE-5, A Program for the Solution of the Multigroup Neutron Diffusion Equations in Two Dimensions with Arbitrary Group Scattering for the Univac 1108 Computer,' GA-8188, Gulf General Atomic Inc. (1967).

(2) R. Froehlich, 'A Theoretical Foundation for Coarse Mesh Variation Techniques,' Proc. Intern. Conf. Res. Reactor Utilization and Reactor Math. Mexico, D.F., 1, 219 (1967).

(3) J. P. Dorsey, 'GAMBLE-4, A Program for the Solution of the Multigroup Neutron Diffusion Equations in Two-Dimensions with Arbitrary Group Scattering for the IBM 7044 Fortran-IV System,' GA-6540, Gulf General Atomic Inc. (1965).

(4) T. B. Fowler, M. Tobias, and D. Vondy, 'Exterminator, a Multigroup Code for Solving Neutron Diffusion Equations in One and Two Dimensions,' ORNL-TM-842, Oak Ridge National Laboratory (1965).

k. Machine Requirements - 65,536 words of core storage, 3 tape units on 1 data channel, 1,572,864 words of FH-880 drum storage from 1 data channel, and a peripheral printer.

l. Programming language used - Fortran IV, but for scratch data handling use is made of Univac 1108 assembly language.

m. Operating System - EXECII, Gax 23.

n. Other programming information -

o. Author of the Abstract - J. P. Dorsey and R. Froehlich
GA Technologies Inc.
P. O. Box 85608
San Diego, CA 92138

SECTION 7.0

OPERATING PROCEDURES

7.0 OPERATING PROCEDURES

The following information provides generic operating procedures. Specific operating instructions with the necessary administrative and quality assurance provisions should be prepared and followed.

7.1 PROCEDURE FOR LOADING THE PACKAGE

7.1.1 Spent Fuel Elements - Model FSV-1 Configuration E

The following procedure is applicable for Model FSV-1 in Configuration E when used for the transport of spent fuel elements from the Fort St. Vrain High Temperature Gas-Cooled Reactor.

Step

1. Moving the Shipping Cask into the Truck Bay.

2. Inspect and clean the tractor, semitrailer and shipping cask as required.

3. Back the tractor and semitrailer into the truck bay.

4. Lower the semitrailer landing gear, disconnect the tractor and drive the tractor out of the truck bay.

5. Preparing the Refueling Floor to Receive the Shipping Cask.

6. Remove the fuel loading port cover.

Step

7. Remove the hatches in refueling floor.

8. Open the sliding hatch above the truck bay.

9. Transferring the Shipping Cask to the Refueling Floor.

10. Loosen the four (4) socket head cap screws that attach the bottom of the shipping cask to the rear support. CAUTION: Do not remove screws.

11. Remove tiedown strap from the front support.

12. Remove nuts and washers to release the retaining ring from the impact limiter.

13. Remove the retaining ring for the impact limiter from the studs.

14. Attach a hoisting sling to the two (2) eyebolts on the impact limiter.

15. Remove the impact limiter from the shipping cask.

16. Engage the cask lifting apparatus in the recessed lifting sockets. Use handcrank to lock the balls in the sockets.

17. Raise the shipping cask approximately twelve (12) inches and slide the retaining ring for the impact limiter to the bottom of the shipping cask.

Step

18. Raise the shipping cask to the vertical position.

    CAUTION: Do not lift the shipping cask while it is still bolted to the trailer.

19. Install the locking block on the rear support structure to hold the trunnion in position.

20. Remove the four (4) socket head cap screws which attach the cask to the rear support.

21. Raise the cask to the refueling floor.

22. Align the arms of the cask lifting apparatus with the guides in the fuel loading port and lower the cask until it is supported by the loading port.

23. Install the bottom restraint on the shipping cask.

24. Disengage the cask lifting apparatus from the cask.

25. Preparing to Load the Shipping Cask.

26. Remove and visually inspect for damage the twenty-four (24) socket head cap screws that hold the outer closure in place. Replace any cap screw found to have stripped or galled threads or any visible deformation of the head or shank. Minor nicks, scrapes or upsets from normal wrench contact are not cause for replacement.

27. Remove the three (3) set screws from the closure and install three (3) eyebolts.

Step

28. Lift the closure from the cask.

29. Examine the seals and sealing surface. Replace the seal if it has nicks, cuts, scratches, or other deformations that could adversely affect seal performance.

30. Remove and examine the twelve (12) inner closure bolts. Replace any bolt found to have stripped or galled threads or any visible deformation of the head or shank. Minor nicks, scrapes or upsets from normal wrench contact are not cause for replacement.

31. Remove the inner closure.

32. Examine the inner closure seal. Replace the seal if it has any nicks, cuts, scratches or other deformations that could adversely affect seal performance.

33. Visually inspect the cavity of the inner container for any damage or debris. If any is noted, evaluate and take corrective action if necessary.

34. Install the inner closure without the twelve bolts.

35. Install the sealing adapter in the top of the shipping cask.

36. Install a reactor isolation valve over the fuel loading port.

37. Install track and adapter guide onto the inner closure. Engage the track on the two keys in the reactor isolation valve.

38. Move the auxiliary transfer cask from storage position to the top of reactor isolation valve located on the fuel loading port.

910013 NC

Step

39. Open the shutter valve of the auxiliary transfer cask and lower the auxiliary transfer cask grapple to engage track and adapter assembly.

40. Raise the inner closure into auxiliary transfer cask.

41. With the purge connection on the reactor isolation valve, purge the auxiliary transfer cask and backfill with air.

42. Close auxiliary transfer cask shutter.

43. Move the auxiliary transfer cask to its storage position and secure.

44. Placing the Spent Fuel Elements in the Shipping Cask.

45. Mount the fuel handling machine containing six (6) spent fuel elements on the reactor isolation valve.

46. Evacuate shipping cask through the reactor isolation valve.

47. Open fuel handling machine cask valve.

48. Operate the fuel handling machine to deposit six (6) spent fuel elements into the shipping cask.

49. Close and seal the fuel handling machine cask valve.

50. Evacuate the shipping cask and backfill with air. Close the reactor isolation valve.

51. Move the fuel handling machine to its storage position.

Step

52. Installing the Container Closure.

53. Move the auxiliary transfer cask from its storage position and mount on the reactor isolation valve over the fuel loading port.

54. Open the reactor isolation valve.

55. Operate the auxiliary transfer cask to lower the inner closure onto the container body.

56. Remove and store the auxiliary transfer cask.

57. Remove track and adapter assembly from the inner closure lid.

58. Install the 12 bolts that attach the inner closure. Torque the bolts to 19-21 ft-lb.

59. Move the reactor isolation valve to its storage position.

60. Leakage Testing the Container Closure Seal.

61. Connect a leakage test system with pressure gauge and shutoff valve to the test port in the container closure.

62. Evacuate seal interspace to a pressure of 1 mm Hg. Disconnect leakage test system from vacuum source.

63. Record the time and abserve any pressure rise for a period of two (2) minutes.* Maximum permissible pressure rise is 5.6 mm Hg. This demonstrates a leakage rate no greater than 1 x 10^{-3} cm^3/sec.

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*Test time is based on a total test cavity volume of 1 cubic in. (16.387 cm^3).

Step

64. Open shutoff valve to vent seal interspace and remove leakage test system.

65. Install the shipping plug in the test port.

66. Remove seal adapter from top of cask.

67. Visually inspect the inner closure for damage or poor fit. If necessary, take corrective action before proceeding further.

68. Preparing the Cask for Shipment.

69. Install the outer closure on the shipping cask.

70. Install the twenty-four (24) outer closure bolts and torque to 570-630 ft-lb.

71. Remove the three (3) eyebolts and install the set screws in the holes.

72. Returning the Cask to the Transport Semitrailer.

73. Engage the cask lifting apparatus in the recessed lifting sockets. Use the handcrank to lock the balls in the sockets.

74. Remove the restraint from the bottom of the cask.

75. Open the sliding hatch above the truck bay.

76. Lift the cask from the fuel loading port, and lower it into the truck bay.

77. Place the retaining ring for the impact limiter on the rear support.

Step

78. Lower the shipping cask onto the rear support. Rotate cask to align the slot in the cask with the key in the rear support.

79. Install the four socket head cap screws which attach the cask to the rear support.

80. Remove the locking block from the rear support structure to allow the trunnion to rotate.

81. Rotate the cask about the rear support trunnion until the cask body is approximately 12 inches above the front support.

82. Slide the retaining ring for the impact limiter to the top of the shipping cask.

83. Lower the cask onto the front support.

84. Release the cask lifting apparatus from the shipping cask and return to storage.

85. Visually inspect the outer closure, center plug and purge cover for significant damage that may affect seal performance. If damage is noted, further examine the component and take corrective action if necessary.

86. Install the tiedown strap at the front support.

87. Attach a hoisting sling to the impact limiter.

88. Install the impact limiter on the shipping cask and attach it with the retaining ring. Torque nuts onto retaining ring stud to 10 ft-lb ± 2 ft-lb.

Step

89. Torque the 4 socket head cap screws that attach the shipping cask to rear support to 522-578 ft-lb.

90. Back the tractor into the truck bay and connect the semitrailer.

91. Drive the tractor with semitrailer from the truck bay.

92. Preparing for Departure.

93. Visually inspect the package. Correct any deficiencies before departure.

94. Survey the shipping cask for external radiation and loose contamination. External radiation and loose contamination shall not exceed the limits of 10CFR71.47 and 10CFR71.87.

95. Attach the proper label to the shipping cask.

96. Display the proper placards on the tractor and semitrailer.

97. Prepare the necessary shipping papers.

98. Dispatch the shipment.

7.1.2. Irradiated Hardware - Model FSV-1 Configurations F and G

The following procedure is applicable for Model FSV-1 in Configurations F and G when used for the transport of solid, nonfissile, irradiated, and contaminated hardware from the Fort St. Vrain High Temperature Gas-Cooled Reactor.

Step

1. Moving the Shipping Cask into the Truck Bay.

2. Inspect and clean the tractor, semitrailer, and shipping cask as required.

3. Back the tractor and semitrailer into the truck bay.

4. Lower the semitrailer landing gear, disconnect the tractor, and drive the tractor out of the truck bay.

5. Preparing the Refueling Floor to Receive the Shipping Cask.

6. Remove the fuel loading port cover.

7. Remove the hatches in refueling floor.

8. Open the sliding hatch above the truck bay.

9. Transferring the Shipping Cask to the Refueling Floor.

Step

10. Loosen the four (4) socket head cap screws that attach the bottom of the shipping cask to the rear support. CAUTION: Do not remove screws.

11. Remove tiedown strap from the front support.

12. Remove nuts and washers to release the retaining ring from the impact limiter.

13. Remove the retaining ring for the impact limiter from the studs.

14. Attach a hoisting sling to the two (2) eyebolts on the impact limiter.

15. Remove the impact limiter from the shipping cask.

16. Engage the cask lifting apparatus in the recessed lifting sockets. Use handcrank to lock the balls in the sockets.

17. Raise the shipping cask approximately twelve (12) inches and slide the retaining ring for the impact limiter to the bottom of the shipping cask.

18. Raise the shipping cask to the vertical position.

    CAUTION: Do not lift the shipping cask while it is still bolted to the trailer.

Step

19. Install the locking block on the rear support structure to hold the trunnion in position.

20. Remove the four (4) socket head cap screws which attach the cask to the rear support.

21. Raise the cask to the refueling floor.

22. Align the arms of the cask lifting apparatus with the guides in the fuel loading port and lower the cask until it is supported by the loading port.

23. Install the bottom restraint on the shipping cask.

24. Disengage the cask lifting apparatus from the cask.

25. Preparing to Load the Shipping Cask.

26. Remove and visually inspect for damage the twenty-four (24) socket head cap screws that hold the outer closure in place. Replace any cap screw found to have stripped or galled threads or any visible deformation of the head or shank. Minor nicks, scrapes or upsets from normal wrench contact are not cause for replacement.

27. Remove the three (3) set screws from the closure and install three (3) eyebolts.

28. Lift the closure from the cask.

29. Examine the seals and sealing surface. Replace the seal if it has nicks, cuts, scratches, or other deformations that will adversely affect seal performance.

Step

30. Visually inspect the cask cavity for any damage or debris. If any is noted, evaluate and take corrective action if necessary.

31. Install the sealing adapter in the top of the shipping cask.

32. Install a reactor isolation valve over the fuel loading port.

33. Loading the Burial Canister into the Shipping Cask.

34. Remove the 22-in.-diameter plug located above the service platform in the hot service facility.

35. Install a reactor isolation valve above the hot service facility.

36. Move the auxiliary service cask to the reactor isolation valve located above the hot service facility.

37. Verify that the handling adapter has been bolted to the top of the burial canister.

38. Operate the auxiliary transfer cask to lift the burial canister up into the transfer cask.

39. Move the auxiliary transfer cask to the reactor isolation valve mounted above the fuel loading port.

40. Operate the auxiliary transfer cask to lower the burial canister into the shipping cask.

41. Move the auxiliary transfer cask to its storage position.

42. Move the reactor isolation valve to a storage position.

Step

43. Remove the sealing adapter.

44. Remove the handling adapter from the top of the burial canister.

45. Rotate the burial canister body and/or the burial canister plug such that the dowel on the retaining ring engages the hole in the plug and the guides on the retaining ring capture the key in the shipping cask body.

46. Install the six (6) socket head cap screws that attach the retaining ring to the burial canister and torque to 79-81 ft-lb.

47. Preparing the Cask for Shipment.

48. Visually inspect the inner closure for damage or poor fit. Take corrective action, if necessary, before proceeding.

49. Install the outer closure on the shipping cask.

50. Install the twenty-four (24) outer closure bolts and torque to 570-630 ft-lb.

51. Remove the three (3) eyebolts and install the set screws in the holes.

52. Remove the plug from the seal test port in the outer closure.

53. Connect the pressure rise leak test system to the test port.

54. Operate the system to evacuate the seal interspace to a pressure of one (1) mm Hg.

Step

55. Close valve and disconnect vacuum pump. Record the time and observe any pressure rise for a period of (2) minutes.* Maximum allowable rise is 5.6 mm Hg. This verifies a leakage rate not greater than 1×10^{-3} atm-cm^3/sec.

56. Disconnect the test system and install the plug in the seal test port.

57. Returning the Cask to the Transport Semitrailer.

58. Engage the cask lifting apparatus in the recessed lifting sockets. Use the handcrank to lock the balls in the sockets.

59. Remove the restraints from the bottom of the cask.

60. Open the sliding hatch above the truck bay.

61. Lift the cask from the fuel loading port, and lower it into the truck bay.

62. Place the retaining ring for the impact limiter on the rear support.

63. Lower the shipping cask onto the rear support. Rotate cask to align the slot in the cask with the key in the rear support.

64. Install the four socket head cap screws which attach the cask to the rear support.

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*Test time is based on a total cavity volume of 1 cubic in. (16.387 cm^3).

Step

65. Remove the locking block from the rear support structure to allow the trunnion to rotate.

66. Rotate the cask about the rear support trunnion until the cask body is approximately 12 inches above the front support.

67. Slide the retaining ring for the impact limiter to the top of the shipping cask.

68. Lower the cask onto the front support.

69. Release the cask lifting apparatus from the shipping cask and return to storage.

70. Install the tiedown strap at the front support.

71. Attach a hoisting sling to the impact limiter.

72. Install the impact limiter on the shipping cask and attach it with the retaining ring. Torque nuts onto retaining ring stud to 10 ft-lb ± 2 ft-lb.

73. Torque the 4 socket head cap screws that attach the shipping cask to rear support to 522-578 ft-lb.

74. Back the tractor into the truck bay and connect the semitrailer.

75. Drive the tractor with semitrailer from the truck bay.

76. <u>Preparing for Departure</u>.

77. Visually inspect the package. Correct any deficiencies before departure.

Step

78. Survey the shipping cask for external radiation and loose contamination. External radiation and loose contamination shall not exceed the limits of 10CFR71.47 and 10CFR71.87.

79. Attach the proper label to the shipping cask.

80. Display the proper placards on the tractor and semitrailer.

81. Prepare the necessary shipping papers.

82. Dispatch the shipment.

7.2 PROCEDURES FOR UNLOADING THE PACKAGE

7.2.1. Spent Fuel Elements – Model FSV-1 Configuration E

The following procedure is applicable for Model FSV-1 Configuration E when used for the transport of spent fuel elements to a storage facility. The specific procedures for receiving and unloading the cask shall be in compliance with 10CFR20.1906.

STEP

1, Receiving the shipping cask.

2. Verify that placards, labels, and shipping papers are in place and correct.

3. Inspect and clean the tractor, semitrailer, and cask as required.

4. Removing the Cask from the Semitrailer.

5. Loosen the four (4) socket head cap screws that attach the bottom of the shipping cask to the rear support on the semitrailer.
CAUTION: Do not remove.

6. Remove tiedown strap from front support.

7. Remove nuts and washers to release the retaining ring from the impact limiter.

8. Remove the retaing ring from the studs.

9. Attach a hoisting sling to the two (2) eyebolts on the impact limiter.

Step

10. Carefully slide the impact limiter from the cask.

11. Attach the cask lifting apparátus to the crane hook.

12. Engage the çask lifting apparatus in the recessed lifting sockets. Use handcrank to lock the balls into the sockets.

13. Raise the cask slightly and slide the retaining ring for the impact limiter to the base of the cask.

14. Raise the cask to the vertical position. Install the locking block on the rear support structure to hold the trunnion in position. CAUTION: Do not lift cask while attached to semitrailer.

15. Remove the four (4) socket head cap crews that attach the bottom of the shipping cask to the rear support.

16. Raise the cask to clear the rear support.

17. Move the cask to the unloading area and lower in the vertical position using a suitable support.

18. Unloading the Shipping Cask.

19. Attach suitable lifting fixture to three (3) threaded holes in the outer closure.

Step

20. Remove the twenty-four (24) outer closure bolts.

21. Remove the outer closure.

    NOTE: Steps 22 through 26 are optional depending on the storage facility requirements.

22. Remove the shipping plug from the seal test port in the inner container closure.

23. Remove the primary plug from the seal test port in the inner container closure.

24. Install the cavity gas sampling adapter in the seal test port.

25. Operate an evacuation system to draw a sample of the cavity gas from the inner container.

26. Disconnect the evacuation system and install the primary plug and the shipping plug.

27. Attach a lifting fixture to the inner closure using the three (3) threaded inserts.

28. Remove the twelve closure bolts.

    CAUTION: The next two steps must be performed remotely.

Step

29. Remove the inner closure.

30. Use a suitable grapple to remove the six (6) spent fuel elements from the inner container.

31. Visually examine the inner container sealing surfaces and interior for any damage or debris.

32. Visually examine the inner closure, especially the Gask-O-Seal assembly for any damage.

33. Visually examine the cask body sealing surfaces for any damage or debris.

34. Install the inner closure and torque the 12 bolts to 19-21 ft-lb.

35. Visually examine the outer closure, especially the Gask-O-Seal assembly for any damage.

36. Install the cask closure and torque the closure bolts to 95-105 ft-lb.

37. Raise the cask in to the vertical position and return to the semitrailer.

38. Position the cask above the bottom support and lower the cask. Retaining ring for impact limiter must be on the rear support.

Step

39. Install the four (4) socket head cap screws which attach the cask to the rear support.

40. Remove the locking block from the rear support structure to allow the trunnion to rotate.

41. Lower the cask until slightly above the front support.

42. Slide the retaining ring for the impact limiter up to the shoulder at the top of the cask.

43. Lower the cask onto the front support.

44. Release the cask lifting apparatus from the cask.

45. Install the tiedown strap at the front support.

46. Attach the hoisting sling to the impact limiter.

47. Slide the impact limiter onto the cask and attach it with the retaining ring. Torque nuts onto retaining ring stud to 10 ft-lb $\pm$ 2 ft-lb.

48. Torque the four (4) socket head cap screws that attach the bottom of the cask to rear support to 522-578 ft-lb.

7.2.2. Irradiated Hardware – Model FSV-1 Configurations F and G

The following procedure is applicable for Model FSV-1 in Configurations F or G when used for the transport of solid, nonfissile, irradiated and contaminated hardware to a disposal site. The specific procedures for receiving and unloading the cask shall be in compliance with 10CFR20.1906.

STEP

1, Receiving the shipping cask.

2. Verify that placards, labels, and shipping papers are in place and correct.

3. Inspect and clean the tractor, semitrailer, and cask as required.

4. Removing the Cask from the Semitrailer.

5. Loosen the four (4) socket head cap screws that attach the bottom of the shipping cask to the rear support on the semitrailer.
CAUTION: Do not remove.

6. Remove tiedown strap from front support.

7. Remove nuts and washers to release the retaining ring from the impact limiter.

8. Remove the retaining ring from the studs.

9. Attach a hoisting sling to the two (2) eyebolts on the impact limiter.

Step

10. Carefully slide the impact limiter from the cask.

11. Attach the cask lifting apparatus to the crane hook.

12. Engage the cask lifting apparatus in the recessed lifting sockets. Use handcrank to lock the balls into the sockets.

13. Raise the cask slightly and slide the retaining ring for the impact limiter to the base of the cask.

14. Raise the cask to the vertical position. Install the locking block on the rear support structure to hold the trunnion in position. CAUTION: Do not lift cask while attached to semitrailer.

15. Remove the four (4) socket head cap crews that attach the bottom of the shipping cask to the rear support.

16. Raise the cask to clear the rear support.

17. Move the cask to the unloading area and lower to the horizontal position on a suitable support.

18. Unloading the Shipping Cask.

19. Attach suitable lifting fixture to three (3) threaded holes in the outer closure.

20. Remove the twenty-four (24) closure bolts.

Step

21. Remove the closure.

22. Attach a hook to the clevis located in the closure plug in the burial canister.

23. Use suitable guide fixture to protect the cask surfaces and slide the burial canister from the cask.

24. Returning the Cask to the Semitrailer.

25. Remove the guide fixture.

26. Visually examine the cask sealing surfaces and cask interior for any damage or debris.

27. Visually examine the cask closure, especially the Gask-O-Seal assembly, for any damage.

28. Install the cask closure and torque the closure bolts to 95-105 ft-lb.

29. Raise the cask in to the vertical position and return to the semitrailer.

30. Position the cask above the rear support and lower the cask. Retaining ring for impact limiter must be on the rear support.

Step

31. Install the four (4) socket head cap screws which attach the cask to the rear support.

32. Remove the locking block from the rear support structure to allow the trunnion to rotate.

33. Lower the cask until slightly above the front support.

34. Slide the retaining ring for the impact limiter up to the shoulder at the top of the cask.

35. Lower the cask onto the front support.

36. Release the cask lifting apparatus from the cask.

37. Install the tiedown strap at the front support.

38. Attach the hoisting sling to the impact limiter.

39. Slide the impact limiter onto the cask and attach it with the retaining ring. Torque nuts onto retaining ring stud to 10 ft-lb ± 2 ft-lb.

40. Torque the four (4) socket head cap screws that attach the bottom of the cask to rear support to 522-578 ft-lb.

7.3 PREPARATION OF AN EMPTY PACKAGE FOR TRANSPORT

The specific procedures for the shipment of an empty cask shall be in compliance with 49CFR173.428.

STEP

1. Survey the shipping cask for loose radioactive contamination and external radiation.

2. Display the proper placards on the tractor and semitrailer.

3. Prepare the necessary shipping papers.

4. Dispatch the shipment.

SECTION 8.0

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 ACCEPTANCE TESTS

There are no current plans to fabricate any additional units of Model FSV-1; therefore, acceptance tests are not required nor provided as a part of this consolidated design report.

8.2 MAINTENANCE PROGRAM

The following information constitutes a generic maintenance program. A specific maintenance plan with the necessary administrative and quality assurance provisions should be prepared and followed.

8.2.1 Maintenance Program for the Model FSV-1 in Configuration E

The maintenance program for Model FSV-1 in Configuration E consists of leak tests and a visual inspection of the entire packaging. These tests and inspection must be accomplished within the twelve (12) months prior to any use of the packaging. Other inspection techniques may be used to verify the results of the visual inspection.

8.2.1.1 Visual Inspections

Step

1. Remove impact limiter from cask.

2. Remove the cask from the transport semitrailer.

Step

3. Visually inspect the external surface of the cask to detect any gouges, dents, or cracks.

4. Visually inspect the impact limiter for any cracks or other damage.

5. Remove the outer closure from the cask and remove the Gask-O-Seal assembly from the closure.

6. Remove the inner container from the cask body.

7. Visually inspect the outer closure for any damage especially the sealing surface. Also inspect closure bolts.

8. Remove the plug from the leak test port in the outer closure.

9. Visually inspect the plug and the test port for damage to the threads or the sealing surfaces.

10. Remove the lifting sockets from the cask body.

11. Visually inspect the lifting sockets for galling or other damage. Also inspect the attach bolts and the threaded holes in the cask body.

12. Remove the purge connection cover from the bottom of the cask.

Step

13. Remove the elastomer and the metal O-rings from the purge connection cover and visually inspect the cover for damage, especially the O-ring grooves.

14. Remove the quick disconnect fitting and visually inspect for damage.

15. Visually inspect the three purge connection cover attach bolts and the threaded holes in the cask body. Inspect the sealing surface for the O-rings in the cask body.

16. Remove the center plug from the bottom of the cask.

17. Remove the elastomer and the metal O-ring from the center plug and visually inspect the plug for damage, especially the O-ring grooves.

18. Visually inspect the four center plug attach bolts and the threaded holes in the cask body. Inspect the sealing surface for the O-rings in the cask body.

19. Visually inspect the four threaded inserts in the bottom of the cask and the four socket head cap screws used to attach the cask to the rear support on the semitrailer.

20. Visually inspect the interior of the cask body, especially the sealing surface at the open end.

Step

21. Visually inspect the twenty-four (24) closure attach bolts and the twenty-four (24) threaded inserts in the cask body.

22. Visually inspect the external surface of the inner container to detect any gouges, dents, cracks, or other damage.

23. Remove the closure from the inner container body and remove the Gask-O-Seal assembly from the closure.

24. Remove the shipping plug from the seal test port in the inner container closure and visually inspect threads and sealing surfaces for damage.

25. Remove the primary plug from the seal test port in the inner container closure and visually inspect the threads and sealing surfaces for damage.

26. Visually inspect the inner closure for any damage especially the sealing surface.

27. Visually inspect the twelve (12) closure attach bolts and the twelve (12) threaded inserts in the inner container body.

28. Visually inspect the interior of the inner container body, especially the sealing surface at the open end.

8.1.2 Leak Tests

Step

1. Install a new Gask-O-Seal assembly on the inner container closure. A light film of vacuum grease may be applied.

2. Install a new elastomer O-ring seal on the shipping plug. A light film of vacuum grease may be applied.

3. Install a new elastomer O-ring seal on the primary plug. A light film of vacuum grease may be applied.

4. Install the inner container closure and torque the twelve (12) closure bolts to 19-21 ft-lb.

5. Install the cavity gas sampling adapter in the seal test port and pressurize the inner container to 15 psig using dry air or other gas.

6. Disconnect the supply and observe the pressure for at least 30 minutes using a pressure gage with 0.5 psig divisions. No decrease in pressure is allowed.

7. Depressurize the inner container and remove the cavity gas sampling adapter.

8. Install the primary plug in the seal test port.

9. Connect the pressure rise test equipment to the seal test port.

Step

10. Operate the test equipment to evacuate the seal interspace to a pressure of one (1) mm Hg. Disconnect the vacuum system, record the time, and observe any pressure rise for a period of five (5) minutes. The maximum allowable pressure rise if 5.7 mm Hg. This demonstrates a leakage rate no greater than 4×10^{-4} cm^3/sec. Note: The allowable pressure rise is based on a total system volume of 16.4 cubic centimeters or one (1) cubic inch.

11. Remove the pressure rise test equipment and install the shipping plug in the seal test port.

12. Install the inner container in the cask body.

13. Install a new Gask-O-Seal assembly on the cask closure. A light film of vacuum grease may be applied.

14. Install a new elastomer O-ring seal on the seal test port plug. A light film of vacuum grease may be applied.

15. Install a new elastomer and a new metal O-ring on the purge connection cover. A light film of vacuum grease may be applied.

16. Install a new elastomer and a new metal O-ring on the center plug. A light film of vacuum grease may be applied.

17. Install the cask closure and torque the twenty-four closure bolt to 570-630 ft-lb.

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Step

18. Install the center plug in the bottom of the cask and torque the four attach bolts to 38-42 ft-lb.

19. Install the purge connection cover and torque the three attach bolts to 19-21 ft-lb.

20. Install the lifting sockets in the cask body. Torque the two retaining bolts to 19-21 ft-lb.

21. Return the cask to the transport semitrailer.

22. Install the impact limiter on the cask. Torque nuts onto retaining ring stud to 8-12 ft-lb.

8.2.2 Maintenance Program for the Model FSV-1 in Configurations F and G

The maintenance program for Model FSV-1 in Configurations F or G consists of leak tests and a visual inspection of the entire packaging. These tests and inspection must be accomplished within the twelve (12) months prior to any use of the packaging. Other inspection techniques may be used to verify the results of the visual inspection.

8.2.2.1 Visual Inspections

Step

1. Remove the impact limiter from the cask.

2. Remove the cask from the transport semitrailer.

Step

3. Visually inspect the external surface of the cask to detect any gouges, dents, or cracks.

4. Visually inspect the impact limiter for any cracks or other damage.

5. Remove the outer closure from the cask body and remove the Gask-O-Seal assembly from the closure.

6. Visually inspect the closure for any damage especially the sealing surface. Also inspect closure bolts.

7. Remove the plug from the leak test port in the closure.

8. Visually inspect the plug and the test port for damage to the threads or the sealing surfaces.

9. Remove the lifting sockets from the cask body.

10. Visually inspect the lifting sockets for galling or other damage. Also inspect the attach bolts and the threaded holes in the cask body.

11. Remove the purge connection cover from the bottom of the cask.

Step

12. Remove the elastomer and the metal O-rings from the purge connection cover and visually inspect the cover for damage, especially the O-ring grooves.

13. Remove the quick disconnect fitting and visually inspect for damage.

14. Visually inspect the three purge connection cover attach bolts and the threaded holes in the cask body. Inspect the sealing surfaces for the O-rings in the cask body.

15. Remove the center plug from the bottom of the cask.

16. Remove the elastomer and the metal O-ring from the center plug and visually inspect the plug for damage, especially the O-ring grooves.

17. Visually inspect the four center plug attach bolts and the threaded holes in the cask body. Inspect the sealing surfaces for the O-rings in the cask body.

18. Visually inspect the four threaded inserts in the bottom of the cask and the four socket head cap screws used to attach the cask to the rear support on the semitrailer.

19. Visually inspect the interior of the cask body, especially the sealing surface at the open end.

Step

20. Visually inspect the twenty-four closure attach bolts and the twenty-four threaded inserts in the cask body.

8.2.2.2 Leak Tests.

1. Install a new Gask-O-Seal assembly on the cask closure. A light film of vacuum grease may be applied.

2. Install a new elastomer O-ring seal on the seal test port plug. A light film of vacuum grease may be applied.

3. Install a new elastomer and a new metal O-ring on the purge connection cover. A light film of vacuum grease may be applied.

4. Install a new elastomer and a new metal O-ring on the center plug. A light film of vacuum grease may be applied.

5. Install the cask closure and torque the twenty-four closure bolts to 570-630 ft-lb.

6. Install the center plug in the bottom of the cask and torque the four attach bolts to 38-42 ft-lb.

7. Pressurize the cask to 15 psig through the purge connection using dry air or other gas.

Step

8. Disconnect the supply hose and install a pressure gauge with 0.5 psig divisions. Record pressure gage reading and observe for at least 30 minutes. No decrease in pressure is allowed.

9. Use a bubble type leak detector solution such as "Snoop" to investigate for leaks around the closure and around the center plug. No indication of leakage is allowed.

10. Remove the pressure gauge and the attaching hose.

11. Install the purge connection cover and torque the three bolts to 19-21 ft-lb.

12. Use a bubble type leak detector solution such as "Snoop" to investigate for leaks around the purge connection cover. No indication of leakage is allowed.

13. Carefully loosen the twenty-four (24) closure bolts and allow the cask pressure to decrease to ambient.

14. Retorque the twenty-four (24) closure bolts to 570-630 ft-lb.

15. Connect a pressure rise test system to the seal test port in the outer closure.

16. Operate the test system to evacuate the seal interspace to a pressure of one (1) mm Hg. Disconnect the vacuum system, record the time, and observe any pressure rise for a period of two (2) minutes. The maximum allowable pressure rise is 5.6 mm Hg.

This demonstrates a leakage rate no greater than 1 x 10^{-3} cm^3/sec. Note that the allowable rise is based on a system volume of 16.4 cm^3 or one (1) $in.^3$.

17. Disconnect the pressure rise test system and install the plug in the seal test port.

18. Install the listing sockets in the cask body. Torque the two retaining bolts to 19-21 ft-lb.

19. Return the cask to the transport semitrailer.

20. Install the impact limiter on the cask and attach it with the retaining ring. Torque the nuts onto the retaining ring studs to 8-12 ft-lb.

8.3 REPAIRS

Any discrepancies identified during the visual inspection or the leak test shall be corrected and verified by repeating the appropriate inspection or the procedure. All repair materials and repair procedures shall be the same or equivalent to those used during the original fabrication.

SECTION 9.0

QUALITY ASSURANCE

9.0 QUALITY ASSURANCE

The maintenance, repair, modification and use of Model FSV-1 in Configurations E, F and G will be accomplished in accordance with a quality assurance program which meets the requirements of the program described in Subpart H of Title 10, Code of Federal Regulations, Part 71 (10CFR71).

GENERAL ATOMICS

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