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ATTACHMENT 2

VOLUME I REPLACEMENT PAGES
(NON-PROPRIETARY)

NOTICE AND DISCLAIMER

All revisions of this report through NEDO-10084-3 were prepared by General Electric Company solely for the use of the U.S. Nuclear Regulatory Commission (NRC) in licensing the IF-300 Shipping Cask. General Electric assumes no responsibility or damage, which may result from any other use of the information disclosed in any revision of this report through NEDO-10084-3.

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In 1998, CHEM-NUCLEAR SYSTEMS (CNS) became the principal Licensee holder for the IF-300 shipping cask. CNS is responsible for all changes to this report starting with NEDO-10084-5.

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APPENDIX D -

I. Introduction

1.1 Purpose

Prior to 1988, General Electric was the principle Licensee holder for the IF-300 shipping cask. From 1988 to 1998, VECTRA was the principle Licensee holder for this system. In 1998, Chem-Nuclear Systems became the principle licensee holder for this system. All references to General Electric as the principle Licensee holder in this report should currently be understood to be referring to CHEM-NUCLEAR SYSTEMS (CNS).

This consolidated Safety Analysis Report (CSAR) represents the technical basis for Certificate of Compliance (C of C) Number 9001, including revisions, for the IF-300 shipping cask. This CSAR can be amended by CNS through the submittal of changes and/or additions which must be reviewed and accepted by the United States Nuclear Regulatory Commission (NRC).

Originally, the authorized contents of the IF-300 cask was restricted to Group I type fuel bundles, which included 14 x 14 and 15 x 15 fuel rod arrays for PWR bundles and 7 x 7 fuel rod arrays for BWR bundles.

In 1982, the C of C was amended to authorize shipment of Group II fuel bundles which had begun to replace the Group I fuel bundles in operating reactors. The Group II fuel bundles have a larger number of smaller diameter fuel rods. ~~Group II fuel bundles have a larger number of smaller diameter fuel rods.~~ Group II includes the 16 x 16 and 17 x 17 fuel rod arrays for PWRs and the 8 x 8 fuel rod array for BWRs.

Shipment of solid, non fissile, irradiated hardware was authorized in a 1984 amendment.

Shipments of BWR fuel with channels in a 17 element fuel basket (Volume 3, Appendix A) was authorized in a 1991 amendment and shipments of high burnup PWR fuel (Volume 3, Appendix B) was authorized in a 1994 amendment. The use of outer plastic wrap to contain "weeping" was also authorized in a 1991 amendment (Volume 3, Appendix C).

Shipments of higher burnup and enrichment BWR fuel with channels in a 17 element fuel basket and shipments of higher burnup and enrichment PWR fuel (Volume 3, Appendix D) was authorized in a 1999 amendment.

1.2 IF-300 Cask

The IF-300 cask is designed to meet or exceed all NRC and Department of Transportation (DOT) regulations governing the shipment of radioactive material. The primary transportation mode is by railroad, although the shipping package is designed to facilitate truck shipment on a special overweight basis for short distances. This features allows the servicing of reactor sites and other facilities which lack direct railroad access.

The IF-300 cask body is a depleted uranium shielded, stainless steel clad annular cylinder, closed at one end. Fuel is loaded into the cask through the open end and the cask is closed with a bolted and sealed head. The head construction is similar to the body of the cask.

Fuel bundles are located within the cask cavity by a removable stainless steel basket. There are several basket configurations which may be used, depending on the specific fuel being shipped. There are also two heads which permit a variation in cask cavity length. When solid, nonfissile, irradiated hardware is shipped, it is placed within a non-reusable steel liner liner built specifically for that hardware. The cask cavity is air-filled and utilizes a rupture disk device for over pressure protection.

The cask outer surface has large circumferential fins designed for impact protection. Encircling the active fuel zone is a water-filled annulus with corrugated jacket which acts as a neutron shield. The upper and lower ends of the cask are also equipped with sacrificial fins for impact protection.

The cask is cooled, when desired, by diesel engine driven blowers which maintain outer surfaces at temperatures facilitating handling. The cooling system is not required to preserve cask integrity or retain coolant. Four longitudinal ducts direct air from two blowers onto the corrugated surface.

The cask, cask supports, and cooling system are all mounted on a steel skid. Exclusion from the cask and cooling system is provided by a wire mesh enclosure which is retractable for cask removal and locks in place during transport. The skid mounted equipment forms a completely self-contained irradiated fuel and hardware shipping package.

II. DESIGN SUMMARY

2.1 CASK DESCRIPTION

The CNS IF-300 Spent Fuel Shipping Cask is designed in accordance with the criteria of Federal Regulations 10CFR71 and 49CFR173.

Prior to 1994, the fuel loadings which could be contained in the IF-300 are as follows:

Table II-1

FUEL LOADINGS

Reactor Type	NSSS Manufacturer	No. of Bundles	Fuel Rod Array	Cladding Material	Fuel Group
BWR	General Electric	18	7 x 7	Zircaloy	I
PWR	Westinghouse	7	14 x 14	Stainless Steel	I
	Westinghouse	7	14 x 14	Zircaloy	I
	Westinghouse	7	15 x 15	Stainless Steel	I
	Westinghouse	7	15 x 15	Zircaloy	I
PWR	Combustion Eng'rg	7	15 x 15	Zircaloy	I
PWR	Babcock-Wilcox	7	15 x 15	Zircaloy	I
BWR	General Electric	18	8 x 8	Zircaloy	II
PWR	Westinghouse	7	17 x 17	Zircaloy	II
PWR	Combustion Eng'rg	7	16 x 16	Zircaloy	II
PWR	Babcock-Wilcox	7	17 x 17	Zircaloy	II

Since 1994, the IF-300 cask has also been permitted to contain 15 x 15 PWR fuel with a maximum burnup of 45,000 MWd/MTU with a minimum cooling time of 60 months as described in Volume 3, Appendix B.

Since 1999, the IF-300 cask has also been permitted to contain up to six 15 x 15 PWR fuel assemblies with a maximum enrichment of 4.65 wt% ^{235}U, a maximum burnup of 50,000 MWd/MTU with a minimum cooling time of 60 months as described in Volume 3, Appendix D. The IF-300 cask has also been permitted to contain up to 17 BWR fuel assemblies with a maximum enrichment of 4.25 wt% ^{235}U, a maximum burnup of 45,000 MWd/MTU with a minimum cooling time of 48 months, which is also described in Volume 3, Appendix D.

Either BWR or PWR fuel bundles can be accommodated through the use of removable fuel baskets, spacers and two different length closure heads. In addition to irradiated fuel bundles, the IF-300 cask may be used to transport solid non-fissile irradiated hardware.

The cask weight when loaded is between 130,000 and 140,000 pounds depending on the particular type of fuel being shipped. The skid and cooling system weigh approximately 45,000 pounds.

The cask is mounted horizontally on an equipment skid during transport. Although transportation is primarily by rail, the skid is designed to accept wheel assemblies for short haul, special permit trucking. This dual-mode shipping configuration permits the use of the IF-300 cask at those reactor sites which have no direct rail access.

The cask is supported on the skid by a saddle at the head end and a cradle at the bottom end. The cradle forms the pivot about which the cask is rotated for vertical removal from the skid. There is one pickup position on the cask body just below the closure flange. The support saddle engages the cask at this section. The lifting trunnions are removed during transport. The pivot cradle trunnions are slightly eccentric to ensure the proper rotation direction for cask lay-down. The cradle is counter-weighted to remain horizontal when the cask is removed.

The cask is lifted by one of two special yokes, a normal unit and a redundant unit. Either yoke accepts the reactor building crane hook in its upper end and engages the cask lifting trunnions with its lower end. Each yoke is designed to be used with either head. The cask head is removed using four steel cables which are attached to the lifting yoke. The same yoke is used for cask uprighting and cask lifting.

All external and internal surfaces of the cask are stainless steel. The outer shell of the cask body is CG-8M (317) stainless steel. The inner shell is 317 or 216 stainless steel. The circumferential fins are 216 stainless steel and the flanges and end fins are 304 stainless steel. The fuel baskets are made of 216 and 304 stainless steel.

Gamma and fast neutron shielding, respectively, are provided in the If-300 cask by depleted uranium metal between the cask shells and a water/ethylene glycol mixture filled annulus surrounding them. The thin walled jacket which retains the neutron shielding water is fabricated from stainless steel and is corrugated for maximum strength and heat transfer.

The closure head is sealed with a Grayloc metallic ring. The cavity maximum normal operation pressure (LOMC) is 29 psig. However, the design working pressure is 400 psig and overpressure protection is provided by a rupture disk device designed to have a bursting pressure of 350-400 psig at 443 degrees fahrenheit. The rupture disk device is located in one of two cavity valve boxes.

Each cavity valve box is equipped with one nuclear service fill, drain, and vent valve. For ease in servicing, these valves have a quick disconnect fitting which may, as an option, be replaced by a stainless steel pipe cap or pipe plug during cask shipment.

The neutron shielding annulus is partitioned into two separate sections, each protected from overpressure by a 200 psig relief valve located in one of two neutron shielding valve boxes. Service to each section is provided annually through fill, drain, and vent valves also located in the neutron shielding valve boxes. These valves may be replaced by a stainless steel blind flange. All valve handles and/or blind flange bolts are lockwired during transit to prevent loosening.

A thermocouple well is attached to the outside of the inner shell at a point expected to be at the highest temperature. The thermocouple well emerges from the cask bottom and accepts a replaceable chromelalumel thermocouple.

The fuel bundles are contained within a removable, slotted, stainless steel basket. Criticality control is achieved by using B4C filled, stainless steel tubes installed in the basket. Fuel bundles are restrained axially by spacers mounted on the inside of the closure head or in the bottom of the fuel basket. The basket is centered within the cask cavity by disk spacers. Nine such spacers are mounted along the fuel basket length. Fuel bundles are inserted and removed from the basket using standard grapples. The basket is removed when the cask is to be used for the shipment of another fuel type or for cask cavity cleaning. The BWR basket has stainless steel clad uranium shielding pieces mounted on the end adjacent to the cask flange.

The outer surface of the cask body is finned for impact protection. These fins are stainless steel and are circmuferential to the cask

b. 200 psig Pressure Relief Valve: This valve provides overpressure protection to the neutron shielding cavities.

c. 2-Inch Globe Valve: Valves of this type are used for draining, filling, and venting of the cask inner cavity and, optionally, the neutron shielding cavities.

These components are described in greater detail in Section 6.

2.3.5 Thermal Testing of the Cask

Section 6.8 discusses the details of the thermal test procedures, cask thermal acceptance criteria, and the results of tests on casks 301 through 304. The data obtained from these from these tests was used to determine the maximum permissible wet shipment load for each cask and to "calibrate" the thermal model.

The difference in maximum permissible wet shipment heat load between casks 301 through 304 was less than 10%. For dry shipment all casks are rated at 40,000 Btu/hr. maximum heat load.

In addition to above described beginning-of-life thermal tests, each cask has temperature measurements taken while in use. These measurements are reviewed and evaluated on an annual basis to determine if there has been any degradation in the casks ability to dissipate heat.

2.4 CRITICALITY ANALYSIS

Table II-6 summarizes the most reactive criticality conditions for the reference fuels and the configurations indicated licensed prior to 1991. (Volume 3, Appendices A, B, and D, provide details for fuels licensed since 1991). In both the BWR and PWR cases, the use of criticality control members is necessary. For fuels licensed prior to 1991, these are in the form of boron carbide-filled stainless steel tubes (as opposed to borated stainless steel poison plates described in Volume 3, Appendix A) fixed to the fuel basket components. These rods are patterned after the BWR control blade elements. Boron density is 1.75 gm/cc. The poison locations are shown in Section VII.

Table II-6

MAXIMUM K VALUES

Fuel Type	No. of Bundles	Enrichment w/o U^{235}	K
BWR	18	4.0	0.880
PWR	7	4.0	0.955

An infinite array of casks in air with no spacing between the casks raises the K a very small amount thus classifying the cask fissile Class I.

Prior to making a determination of cask k-effective (K) it was necessary to compute the most reactive fuel bundle geometry. This was done by varying rod pitch within the confines of the corresponding basket channel. To determine maximum cask k the peak bundle geometries were placed in the appropriate cask array models and k was computed as a function of cask cooling termperature. Peak cask reactivity is at 20°C.

2.5 SHIELDING ANALYSIS

The analysis only considers the case of 7 PWR bundles with an exposure of 35 GWD/T, a specific power of 40 kw/kg, and a 120-day cooling time. This represents a "worse" case loading of any reference fuel licensed prior to 1991 for which the cask was originally designed. Both gamma and fast neutron radiation must be considered in designing a shipping package for high exposure light water moderated reactor fuels. Volume III Appendix A, B, and D contain shielding analyses for shipment of fuel licensed since 1991.

2.5.1 Gamma Shielding

The gamma source arises from the decay of the radioisotopes created from the fission process during reactor operation. The source strength is a function of specific power, operating time, and cooling time. Section VIII describes the source term in detail for fuels licensed prior to 1991. Volume 3, Appendices A, B,and D, provide details for fuels licensed since 1991. Depleted uranium metal is the principal gamma shield in the IF-300 cask, although there is a significant contribution from the stainless steel inner and outer shells. The uranium is an annular casting four inches thick clad in stainless steel, and forms the cask body. Head and shielding is accomplished with three inches of stainless-clad uranium. The bottom end requires three and three-quarters of an inch of uranium.

2.5.3 Calculational Results

49CFR173 prescribes the allowable dose rates as 10 mr/hr total radiation at a point 6 feet from the vertical projection of the outer edges of the transport vehicle. Furthermore, DOT and NRC regulations specify a limit of 1 R/hr three feet from the cask surface following the hypothetical accident conditions. Table II-7 indicates that the IF-300 cask shielding meets both normal and accident shielding requirements.

2.6 FISSION PRODUCT RELEASE

For the reduced heat load of dry shipments in the IF-300 cask, the analyses ofSection VI show that there is no release of any of the cask contents to the environs for either normal or accident conditions.

2.7 REGULATIONS

The IF-300 irradiated fuel shipping cask is designed to meet both the normal transport and accident conditions of the NRC and DOT. Section IX summarizes the design results in light of these regulatory criteria.

2.8 OPERATION AND MAINTENANCE

2.8.1 Operation

A complete operating manual has been written and is provided to each cask user. In addition, CNS offers training on cask handling prior to use. CNS-supplied technical assistance will also be offered in support of cask handling. The user is expected to provide all operating and health physics personnel. The user will bear the responsibility of proper cask operations.

Each shipping package will have in-transit instructions which provides guidance to the carrier personnel in the event of an abnormal condition.

2.8.2 Maintenance

Maintenance and repair of the IF-300 cask will be performed following written instructions. The same level of quality specified for the initial fabrication will be applied to maintenance and repair items. Where applicable, manufacturer's recommendations or accepted industry standards will be followed. Records will be maintained on a cask-by-cask basis in accordance with regulatory requirements. An approved quality assurance plan will be applied to all items of maintenance and repair.

2.9 FABRICATION AND QUALITY ASSURANCE

Since it is necessary to have some uniform and familiar set of criteria to govern equipment fabrication, General Electric has chosen the ASME Nuclear Vessel Code, Section III as guidance for the IF-300 cask fabrication and quality control. The design portion of Section III is excluded, due to the unique requirements of shipping casks.

All IF-300 basic components identified in Chapter IX are designed, fabricated, tested, used, and maintained under an NRC approved quality assurance program that satisfies requirements in 10CFR71 Subpart H "Quality Assurance" criteria for packaging and transportation of radioactive material.

III. FUELS AND CONTENTS DESCRIPTION

3.1 INTRODUCTION

The IF-300 spent fuel cask is designed as a general purpose shipping container. With its various length heads and removable fuel baskets, the IF-300 is capable of servicing all of the present and planned light-water moderated power reactors which have a building crane capacity of greater than 70 tons.

The fuels are segregated into two generic groups, BWR and PWR. Within each group is a design basis or reference fuel bundle. This assembly is a composite of parameters based on the present and projected fuel designs of General Electric, Westinghouse, Babcock & Wilcox, and expected values for present generation power reactors. These critical parameters include exposure, specific operating power, enrichment critical parameters include exposure, specific operating power enrichment, uranium content, active length, and bundle cross-section geometry. All of these are necessary inputs to a shipping cask design analysis.

Prior to 1991, the IF-300 cask was designed to ship either eighteen (18) of the BWR reference fuel bundles or seven (7) of the PWR reference fuel bundles. Since 1991, the IF-300 cask has also been permitted to ship seventeen (17) channelled BWR fuel bundles. This approach permits the shipment of any BWR or PWR fuel without specific analysis as long as it is within its respective design basis envelope.

The reference fuels and their bases licensed prior to 1991 are summarized in Table III-1. Fuel designs licensed in 1991 for the channelled BWR fuel basket are summarized in Volume 3, Appendix A, page A-1-9, Table A-1.2-2. PWR fuel with a maximum burnup of 45,000 MWD/MTU licensed in 1994 are described in Volume 3, Appendix B.

PWR fuel with a maximum enrichment of 4.65 wt% ^{235}U, a maximum burnup of 50,000 MWd/MTU and a minimum cooling time of 60 months licensed in 1999 are described in Volume 3, Appendix D. BWR fuel with a maximum enrichment of 4.25 wt% ^{235}U, a maximum burnup of 45,000 MWd/MTU and a minimum cooling time of 48 months licensed in 1999 are also described in Volume 3, Appendix D.

3.2 Fission Product Activities and Powers for the Design Basis Fuels[1]

This section addresses Design Basis fuel licensed prior to 1991. Volume 3, Appendices A and B address fuel licensed for a 17-cell channelled BWR fuel basket in 1991 and 15x15 Westinghouse PWR fuel licensed with a maximum burnup of up to 45,000 MWd/MTU with a minimum cooling time of 60 months in 1994, respectively.

Volume 3, Appendix D addresses fuel licensed for a 17-cell channelled BWR fuel basket with a maximum enrichment of 4.25 wt% ^{235}U, a maximum burnup of 45,000 MWd/MTU and a minimum cooling time of 48 months. Volume 3, Appendix D also addresses 15x15 Westinghouse PWR fuel licensed for a loading of 6 assemblies in the PWR basket with a with a maximum enrichment of 4.65 wt% ^{235}U, a maximum burnup of 50,000 MWd/MTU and a minimum cooling time of 60 months.

3.2.1 Design basis for PWR bundles

3.2.1.1 Determination of the thermal Neutron Flux:

As a basis for design, the specific power is taken to be 40 kw/kgU, and the burn-up to be 35,000 MWd/MTU. The irradiation time is:

[1] Calculations by Dr. Charles B. Magee; Denver Research Institute, Denver Colorado. Results of calculation used in Table III-1.

weight of the package contents , and a horizontal component in the transverse direction of five times the weight of the package and contents.

5.2.10 The cask body shall withstand the thermal stress conditions arising from : 1) normal cooling; 2) loss-of mechanical cooling: 3) partial loss-of shielding water; 4) 30-minute fire; and 5) post-fire equilibrium.

5.3 MATERIALS

Table V-1 presents the materials used in the cask, the 7-cell PWR and 18-cell BWR fuel assembly support baskets licensed prior to 1991, and miscellaneous attachments. Volume 3, Appendix A presents the materials used in the 17-cell channelled BWR fuel assembly support basket licensed in 1991.

5.3.1 Uranium Shielding Specification

The depleted uranium metal shielding material is in the form of annular castings, shrink-fitted together to form a continuous shield for the length of the cask. All casting, handling, testing and preparation for shipment are performed in accordance with General Electric Company approved specifications.

The cast material has a maximum U-235 content of 0.22%. The U-235 content of UF_6 tail material is nominally 0.20% with a $\pm$ 0.02% variation. Isotopic analysis has been performed on each casting to assure compliance with the aforementioned limit. Certified copies of the various analyses were originally provided to General Electric Company and are currently retained by CNS.

Table V-1

MATERIALS

Item	Materials
External water jacket	ASTM A240 Type 304
Inner Shell	ASTM A296-65 CG-8M (317SST modified) or AISI 200 Type 216SST rolled plate
Shielding (casting)	Uranium, depleted metal
Outer body shell	ASTM A296-65 CG-8M (317SST modified)
Structural rings	AISI 200 Type 216
Valve box sides (castings)	ASTM A351-CF* (304SST)
Valve box cover	AISI 200 Type 216
Bottom head outside shell	ASTM A240 Type 304
Bottom head inside shell	ASTM A240 Type 304
Top head outside shell	ASTM A240 Type 304
Top head inside shell	ASTM A240 Type 304
Top head flange (forging)	ASTM A182 304SST
Cask Body Flange (forging)	ASTM A182 304SST
Top head fins	ASTM A240 Type 304
Bottom head fins	ASTM A240 Type 304
Valve box fins	AISI 200 Type 216
Studs and nuts	17-4 PH H-1075/H-1025
Fuel element basket axial supports	AISI 200 Type 216
Fuel element basket channels	ASTM A240 Type 304
Basket support rings	AISI 200 Type 216
Support saddle	ASTM A516 Gr 70
Pivot cradle	ASTM A516 Gr 70
Cradle pedestals	ASTM A516 Gr 70
Block pin	AISI 4340 heat treated
Pivot cradle counter weight	Lead
Lifting trunnion blocks	AISI 4340, 304N or nitronic 40 stainless steel forgings
Cooling ducts	6061/3003 Aluminum
Enclosure	6061/6063/3003 aluminum
Skid	Tri-Ten steel

VI. THERMAL ANALYSIS

6.1 INTRODUCTION

This section sescribes thermal analyses of the IF-300 shipping cask. The characteristics of the types of fuels which may be transported in the cask are described in Section 3.0.

The analyses described in this section assume that the IF-300 cask is used in the "dry" shipping mode with a design basis heat load of 40,000 Btu/hr. The per bundle maximum decay heat rates are as follows:

- PWR fuel - 5725 Btu/hr
- BWR fuel - 2225 Btu/hr

Thermal analyses were originally done in 1973 for wet shipments of high heat load Group I BWR and PWR fuels. Additional analyses were completed for the Group I fuels in 1974 for low heat load dry shipments and in early 1980 for high pressure fuel pins. The newer Group II BWR and PWR fuels were analyzed in late 1980 with their results being similar to those of the Group I fuels. Fuel designs licensed since 1991 are presented in Volume 3, Appendices A, B, and D.

Table VI-1 tabulates the characteristics used in the thermal analyses of the Group I 7 x 7 BWR and 15 x 15 PWR fuels (14 x 14 PWR is bounded by the 15 x 15 fuel) and the Group II 8 x 8 BWR and 17 x 17 PWR fuels (16 x 16 PWR is bounded by the 17 x 17 fuel) licensed prior to 1991.

This section contains five "Design Basis Heat Load Conditions, " normal cooling, loss-of-mechanical cooling (LOMC), 50% shielding water loss, 30 minute fire, and post-fire equilibrium (PFE).

The mechanical cooling system is not required by the NRC. This system has been partially or completely removed from all four IF-300 casks. The thermal results shown in this section. The LOMC results replaces all normal cooling results.

Volume 3, Appendix 4, Section A-3.0 has four design basis heat load conditions, normal cooling (NOC), 30 minute fire, 3-hour post fire and post fire equilibrium (PF3). NOC is natural convection in 130° ambient air, which is the same as LOMC in this section.

6.2 PROCEDURES AND CALCULATIONS

6.2.1 Introduction

The thermal analyses described in this section have been, with minor exceptions, calculated by computer. These calculations are based on parameters specified in Table VI-1. This section of the report describes the methodology incorporated in the various computer codes, discusses the bases for the procedures used, and details the calculations performed.

Table VI-1

CHARACTERISTICS OF CASK AND DESIGN BASIS FUELS

CASK

Type	BWR	PWR
Cavity Length, in	180.25	169.50
Cavity Diameter, in	37.5	37.5
Inner Shell Thickness, in	0.5	0.5
Shielding Thickness, in	4.0	4.0
Outer Shell Thickness, in	1.5	1.5
Cask Linear Surface Area, ft^2/ft	39.2	32.2
Cask Length, Excluding Fins, in	192.3	182.1
Shielding Water, lb	4,540	4,540
Cavity Relief Pressure, psig at 443°F	350-400	350-400
Neutron Shielding Relief Pressure, psig	200	200
Maximum Heat Load, Btu/hr - Air Filled	40,000	40,000

FUELS

Type	BWR	PWR
Number of Fuel-Bearing Rods/Buncle - Group I	49 (7 x 7)	208 (15 x 15)
Number of Fuel-Bearing Rods/Buncle - Group II	62 (8 x 8)	264 (17 x 17)
Exposure, GWd/MTU (average)	35.0*	35.0*
Operating Power, kW/KgU	30.0	40.0
Assembly Decay Heat Rate, (max) BTU/hr, Air Filled	2,225	5,725
Assemblies per Cask Load	18	7
Uranium, kgU/Bundle	198	465

* See Volume 3, Appendix B for high burnup PWR fuel. See Volume 3, Appendix D for higher enrichment and burnup PWR and BWR fuel.

VIII. SHIELDING

8.1 FUEL BASES AND SOURCE TERMS

Section III describes the BWR and PWR design basis fuels and Section IV indicates the maximum number of each type which the IF-300 cask will hold as licensed prior to 1991. Appendices A, B, and D describe BWR and PWR fuel licensed since 1991. Considering 18 BWR bundles or 7 PWR bundles, the latter represents the more severe shielding problem because of its higher specific operating power and higher exposure potential due to greater enrichment. For this reason, the IF-300 cask shielding analysis is based on consideration of 7 PWR design basis bundles. Volume 3, Appendix A describes the shielding analysis for the IF-300 cask with 17 channelled BWR fuel assemblies. Volume 3, Appendix B describes the shielding analysis for the IF-300 cask with higher burnup PWR fuel assemblies. Volume 3, Appendix D describes the shielding analysis for the IF-300 cask with 17 channelled higher enrichment and burnup BWR fuel assemblies, and 6 higher burnup and enrichment PWR assemblies. Table VIII-1 gives the parameters of both reference fuel loadings for comparison. The source term has two components, gamma and fast neutron.

8.1.1 Gamma Radiation

Refer to Volume 3, Appendix V-5 for high burn-up PWR fuel analysis. The gamma source comes from the decay of radioisotopes produced in the fuel during reactor operation. The gamma source strength is a function of fuel operating specific power, irradiation time and cooling time. Table VIII-2 shows a seven group distribution for fuels licensed prior to 1991 (see Volume 3, Appendices A, B, and D for fuels licensed since 1991). The seven group distribution is based on 875 operating days at a specific power of 40 kW/kgU, followed by 120 days of cooling. This forms the shielding computer solution input.

8.1.2 Fast Neutron Radiation

Recent work indicates that light water reactor fuel with a burnup of greater than 20,000 MWd/T will contain sufficient concentrations of transplutonium isotopes to make neutron shielding in a shipping cask a necessity.

TABLE VIII-1

IRRADIATED FUEL PARAMETERS

PWR Parameters	
Specific Power	= 40 kWth/kgU
U/Assembly	= 465 kg
Average Power/Assembly	= 18.28 MWth
Peaking Factor	= 1.2
Peak Power/Assembly	= 21.94 MWth
Power/Basket*	= 153.6 MWth
Vol of 7 bundles	= 1.178×10^6 cm^3
BWR Parameters	
Specific Power	= 30 kWth/kgU
kgU/Assembly	= 198 kgu
Average Power/Assembly	= 5.85 MWth
Peaking Factor	= 1.2
Peak Power/Assembly	= 7.02 MWth
Power/Basket*	= 126.36 MWth
Vol of 18 bundles	= 1.616×10^6 cm^3

* Denotes power of fuel while in reactor

TABLE VIII-2

ENERGY GROUPS

Group	Energy Range	Effective Energy	MEV/Fission
I	>2.6 MeV	2.8 MeV	- NEG -
II	2.2 - 2.6	2.38	1.54×10^{-5}
III	1.8 - 2.2	1.97	4.22×10^{-4}
IV	1.35 - 1.80	1.54	2.42×10^{-4}
V	0.9 - 1.35	1.30	1.08×10^{-4}
VI	0.4 - 0.9	0.80	4.16×10^{-2}
VII	0.1 - 0.4	0.40	6.02×10^{-4}

The seven-group distribution is taken from data published by K. Shure in WAPD-BT-24.

The isotopes that form the primary neutron source in high exposure fuel are Curium 242 and Curium 244. In a U-235 fueled reactor, the formation of one atom of CM-242 requires four neutron capture events, while Cm-244 requires six neutron captures. Thus the concentration of these isotopes will depend, roughly on the fuel exposure to the fourth and to the sixth power until the concentrations approach their

IX. Safety Compliance

9.1 INTRODUCTION

This section is designed to recap and summate this report in light of the requirements of 10CFR71 - Subpart C, and 49CFR173.

9.2 10CFR71

9.2.1 General Standards for All Packaging

9.2.1.1 No Internal Reactions

The cask surfaces and the fuel basket are stainless steel. This material does not react with steam or water either chemically or galvanically. The fuel is designed to be nonreactive in waterfilled systems. The uranium shield is totally clad in stainless steel. A copper diffusion barrier separates the stainless steel from the uranium to prevent the formation of an alloy under high temperature conditions. The entire shipping package is chemically and galvanically inert.

9.2.1.2 Positive Closure

The IF-300 cask head is held in place by 32 bolted studs. The mating flanges are designed to accept a Grayloc metallic gasket with a minimum design pressure of 600 psi. Shear steps are provided in the flange to prevent damage to the gasket under impact. Two tapered guide pins ensure proper head alignment during installation.

9.2.1.3 Lifting Devices

The analysis of Section V indicates that the lifting structures of both the cask and the lid are capable of supporting three times their respective weights without generating stresses in excess of their yield strengths (FS > 1.0). The cask design is such that there are no possible lifting points other than those intended. In addition, the failure of any of the intended lifting structures will not result in a redistribution of shielding or a loss of cask integrity.

9.2.1.4 Tie Down Devices

Section V shows that both the front and rear cask supports are capable of sustaining the combined 10 g longitudinal, 5 g transverse and 2 g vertical forces without generating stresses in excess of their yield strengths (FS > 1.0).

The cask is designed to have only one tiedown method. The failure of either, or both, supports will not impair the ability of the package to meet other requirements. There will be no shielding redistribution or loss of cask integrity.

9.2.2 Structural Standards for Large Quantity Shipping

9.2.2.1 Load Resistance

With the package considered as a simple beam loaded with five times its own weight, the cask body outer shell safety factors in shear and bending are 20.4 and 8.6 respectively, based on allowable stresses.

9.2.2.2 External Pressure

When subjected to an external pressure of 25 psig, the package outer shell safety factors in elastic stability and axial failure exceed unity, based on allowable stresses.

9.2.3 Criticality Standards for Fissile Material Packages

This section addresses the 7-cell PWR and 18-cell BWR fuel baskets licensed prior to 1991. Volume 3, Appendices A, B, and D address fuels and baskets licensed since 1991.

9.2.3.1 Maximum Credible Configuration

Fuel element spacing is provided by the stainless steel basket. The stress analysis of Section V shows that during accident conditions there is no redistribution of fuel. The normal transport arrangement is the maximum credible configuration.

ATTACHMENT 3

APPENDIX D, VOLUME IV (NON-PROPRIETARY)

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D-1.0 INTRODUCTION

Carolina Power and Light (CP&L) Company and Chem-Nuclear Systems, Incorporated, (CNSI) submitted an addendum (reference) to the IF-300 Package Certificate of Compliance (C of C) No. 9001 (C of C No. 9001) to permit transportation of PWR and BWR fuel assemblies with burnup and enrichment values greater than those currently permitted under the C of C. The application was made to support spent fuel transportation from the CP&L Company's H. B. Robinson and Brunswick plants. The fuel assembly types for which the addendum is applicable are 15x15 Siemens Power Corporation (SPC) PWR fuel assemblies for Westinghouse 15x15 class reactors with a maximum burnup of up to 50 GWD/MTU and enrichment of 4.65 wt% ^{235}U, and General Electric BWR fuel assemblies with a maximum burnup of up to 45 GWD/MTU and enrichment of 4.25 wt% ^{235}U. The 50 GWD/MTU burnup and 4.65 wt% ^{235}U enrichment of the 15x15 SPC PWR fuel assemblies exceeded the C of C No. 9001 licensed limits of 45 GWD/MTU and 4 wt% ^{235}U, and the 45 GWD/MTU and enrichment of 4.25 wt% ^{235}U of the GE BWR fuel assemblies exceeded the licensed limit of 35 GWD/MTU and 4 wt% ^{235}U. Table D-1.1 lists the characteristics of the fuel assemblies proposed for the application to the IF-300 C of C based upon the fuel characteristics from the H. B. Robinson and Brunswick plants. The addendum (ref) was incorporated by the USNRC into C of C No. 9001 in 1999.

This appendix presents the calculation of the neutron and gamma source strengths, decay heat, and releasable source terms during hypothetical accident conditions for the higher burnup and enrichment fuel assemblies described in Table D-1.1. It also presents the criticality evaluation for these assemblies. This appendix demonstrates that the IF-300 shipping cask loaded with six 15x15 SPC PWR fuel assemblies or 17 channeled General Electric BWR fuel with the burnup and enrichments listed in Table D-1.1 satisfy all of the transportation requirements specified in 10 CFR 71, as well as the limits specified in the IF-300 C of C (C of C No. 9001) and the Consolidated Safety Analysis Report (CSAR 1995). Note that the center location in the PWR basket must be left empty. If an assembly is loaded in the center PWR basket location and a peripheral location is instead left empty, the cask will not meet the criticality safety criteria with 6 assemblies at 4.65 wt% ^{235}U.

Table D-1.1

Characteristics of Proposed Assemblies

Parameter	PWR Fuel	BWR Fuel
Fuel Type	15x15 SPC PWR for Westinghouse Class 15x15 Reactors	GE-7, 8, 9, 10 & 13 BWR Fuel for General Electric BWR/4 Plant Design
Uranium Weight	437 kg/assembly	187 kg/assembly
Number of Assemblies	6[a]	17 channeled
Maximum Assembly Average Burnup	50 GWD/MTU	45 GWD/MTU
Maximum Lattice Average Enrichment	4.65 wt% ^{235}U	4.25 wt% ^{235}U
Minimum Cooling Time	5 years	4 years

[a] The center location in the PWR basket must be left empty. If an assembly is loaded in the center PWR basket location and a peripheral location is instead left empty, the cask will not meet the criticality safety criteria with 6 assemblies at 4.65 wt% ^{235}U.

D-2.0 ANALYSIS METHODOLOGY

The approach used in this appendix to address the effect of the change on the pre-1995 IF-300 certification is described in this section.

D-2.1 Structural Assessment

The requested change deals with increasing the permissible burnup and enrichment of fuel to be shipped in the IF-300 package for specific types of PWR and BWR fuel, which are designated as Group III fuel assemblies. The Group III fuel assemblies have outer dimension envelopes that are compatible with the Group I and II fuel designs, and uranium loadings that are bounded by the Group I and II fuel designs already permitted under the IF-300 Cask C of C (C of C No. 9001). Therefore, there is no change in the structural aspects of the contents to be shipped. Furthermore, since Section 3.0 indicates that the number of moles of residual gas available for release in the cask cavity and the total decay heat load will remain within the pre-1994 licensed limit of 0.5 moles and 40,000 Btu/h, respectively, no change in the internal pressure will occur. Therefore, the proposed change has no effect on the structural analysis of the cask upon which the current C of C (C of C No. 9001) is based.

D-2.2 Thermal Assessment

As discussed above, the pre-1994 licensed total cask inner cavity heat load capacity of the cask of 40,000 Btu/h will be maintained for the higher burnup fuel. As such, it is necessary to determine the required cooling time for the higher burned fuel assemblies to remain within the licensed thermal load limit. The evaluation presented in Section D-3.0 of this appendix indicates that a minimum cooling time of 5 years is required for an IF-300 cask loaded with six SPC PWR fuel assemblies with a burnup of up to 50 GWD/MTU and enrichment from 3.45 to 4.65 wt% ^{235}U for the total cask inner cavity heat load capacity of the cask to be less than 40,000 Btu/h. A minimum cooling time of 4 years is required for an IF-300 cask loaded with seventeen channeled GE BWR fuel assemblies with a burnup of up to 45 GWD/MTU and enrichment from 3.19 to 4.25 wt% ^{235}U for the total cask inner cavity heat load capacity of the cask to be less than 40,000 Btu/h.

D-2.3 Containment Assessment

The containment assessment includes an evaluation of the releasable source terms and fission product gas generation. The releasable source terms include activation products ("crud") adhering to the exterior surfaces of the fuel assemblies and the fission gas products (mainly ^{85}Kr). For higher burnup and enrichment fuel it is anticipated that the releasable source term will be greater than those for lower burnup fuel assemblies, however, the minimum cooling time is increased to offset this increase. As such, it is necessary to calculate the releasable source terms and compare them to the values used in the previous revision of the IF-300 CSAR (CSAR 1995) as the licensing basis. The evaluation of releasable source terms is provided in Section D-7.0 of this appendix. The calculations of the crud source term is described in Section D-4.0. The calculation of the fission product gas generation is described in Section D-3.5.

D-2.4 Shielding Assessment

It is expected that the radioactive source terms will be higher for the higher burnup fuel, however, the minimum cooling time is increased to offset this increase. Therefore, the following parameters are reviewed to assure that the proposed fuel assemblies are equivalent to or less limiting than the fuel assemblies authorized prior to 1995 for shipment in the IF-300 Package, or that the resulting dose rates remain within the allowable transportation dose rate limits:

- Gamma Source Term
- Neutron Source Term
- Hardware Activation

The calculations of these parameters are described in detail in Section D-3.0.

D-2.5 Criticality Assessment

The requested change deals with increasing the permissible burnup and enrichment of fuel to be shipped in the IF-300 package for a specific type of PWR and BWR fuel. Section D-6.0 contains a criticality evaluation for the proposed fuel types. The evaluation demonstrates that the IF-300 Cask with the contents described in Table D-1.1 meets all of the criticality safety criteria contained in 10 CFR 71.

D-3.0 NEUTRON, GAMMA, AND FISSION GAS SOURCE STRENGTH AND DECAY HEAT CALCUATION

This section determines the neutron and gamma source strengths and the decay heat for the higher burnup and higher enrichment fuel assemblies described in Table D-1.1. Determination of fission gases which also contribute to the releasable source term are also calculated. The calculation of the neutron and gamma source strengths are performed using ORIGEN2 (Savino 1999). ORIGEN2 is a computer code which calculates the buildup and decay of radioactive materials. ORIGEN2 code was developed by the Oak Ridge National Laboratory and is distributed through the Radiation Shielding Information Computational Center.

Sections D-3.1 and D-3.2 describe the ORIGEN2 models for the proposed PWR and BWR fuel assemblies, respectively. Sections D-3.3 and D-3.4 describe the results of the ORIGEN2 calculations for the proposed PWR and BWR fuel assemblies, respectively.

D-3.1 ORIGEN2 Input File Preparation for SPC 15x15 PWR Fuel

The SPC 15X15 PWR fuel assembly is assumed to consist of four axial regions for the source term calculation: the active fuel region, the gas plenum region, the upper tie plate region, and the lower tie plate region.

The SPC 15X15 PWR fuel assembly consists of 204 fueled rods, 20 control rod guide tubes, and one instrument tube. The fuel rod cladding material is Zircaloy-4. The cladding extends into the top plenum zone and the bottom end fitting zone. The total rod length is [redacted] in out of which 144 in is in the active fuel zone, [redacted] in is in the top plenum zone, and [redacted] in is in the lower tie plate zone. The Zircaloy-4 weight due to cladding in various regions of the fuel assembly is assumed to be proportional to the cladding lengths in these regions. The component weights were taken from Kunita (1998) and are described below.

The highlighted data is proprietary information to Siemens Power Corporation.

ORIGEN2 models were developed for the maximum and minimum lattice average enrichment ranges which are 4.65 wt% and 3.45 wt% ^{235}U, respectively. The models for the two enrichments are identical with the exception of the uranium loading for the active fuel region.

D-3.1.1 Active Fuel Region

The 144 in long active fuel region includes the following components:

- Fuel - UO_2
- Cladding (204 rods) - Zircaloy-4
- Guide Tubes (20 rods) - Zircaloy-4
- Instrument Tube (1 rod) - Zircaloy-4
- Spacers- Incore - Zircaloy-4/Inconel-718
- Spacer Sleeves- Zircaloy-4

The weight of heavy metal (Uranium) is assumed to be [redacted] kg (Kunita 1998). The total weight of the zircaloy in the active fuel region is 133.927 kg. The weight of Inconel-718 in the incore spacers is 0.112 kg.

D-3.1.2 Upper Tie Plate Region

The upper tie plate region consists of the upper tie plate and the upper locking hardware:

- Upper Tie Plate - 304 stainless steel
- Spring Clamps - 304 stainless steel
- Locking Bars - 304 stainless steel
- Guide Tube Washers - 304 stainless steel
- Locking Springs - Inconel X-750
- Leaf Springs/Supports - Inconel 718
- Locking Lugs/Rings - Inconel 718

The highlighted data is proprietary information to Siemens Power Corporation.

All the steel in the upper tie plate region is modeled as SS-304. The total weight of the steel in the upper tie plate region is 6.381 kg. The weight of the Inconel-X750 is 0.024 kg and the Inconel 718 is 672 g.

D-3.1.3 Gas Plenum Region

The plenum region consists of the following components:

- Cladding (204 rods) - Zircaloy-4
- Guide Tubes (20 rods) - Zircaloy-4
- Instrument Tube (1 rod) - Zircaloy-4
- Sleeves- Zircaloy-4
- Plenum Springs - Inconel-X750

The total weight of zircaloy in the plenum region is 7.551 kg and the weight of the Inconel X-750 plenum-springs is 3.019 kg (Kunita 1998).

D-3.1.4 Lower Tie Plate Region

The lower tie plate region consists of the following components:

- Bottom Tie Plate - 304 stainless steel
- Guide Tube Screws - 304 stainless steel

The total weight of the stainless steel in the lower tie plate region is 4.459 kg.

A summary of the material weights in each region is given in Table D-3.1. These material weights are used in the ORIGEN2 analysis. The elemental composition for each of the materials was taken from DOE (1992).

D-3.1.5 Modeling Bases and Assumptions

The following lists the modeling bases and assumptions used in the generation of the source terms for the SPC PWR fuel. Appendix D-9.1 provides input and output files for the ORIGEN2 cases. Section D-5.0 contains a comparison between the results of these ORIGEN2 runs and the current design basis values for the IF-300 cask.

1. The maximum and minimum lattice average enrichment for the 15x15 SPC PWR fuel from Robinson are 4.65 wt% ^{235}U and 3.45 wt% ^{235}U, respectively.

2. The maximum burnup for the PWR fuel is 50 GWD/MTU.

3. The IF-300 Cask with the PWR basket will be loaded with 6 fuel assemblies, although the basket is capable of holding 7 fuel assemblies. The cask loading is limited to 6 fuel assemblies due to criticality safety concerns.

4. The operating history for the fuel assemblies is assumed to be a high burnup three cycle history similar to the one used in the OCRWM database (DOE 1992).

5. A conservative specific power of 40 MW/MTU during irradiation is used.

6. The fuel assembly material compositions are taken directly from data supplied by CP&L (Kunita 1998).

7. The material weights of burnable poison materials in the higher burnup fuel assembly is not considered in the ORIGEN2 run. This has a very small impact on the overall source strength and decay heat because of their relatively small weights compared to other materials of the fuel assembly.

8. During the irradiation period in the ORIGEN2 model, the fuel assembly is divided into four regions, which are defined as the active fuel region (the portion of the fuel rods associated with the active fuel), the gas plenum region (the portion of the fuel rods above the active fuel length), the upper tie plate region (the remainder of the assembly above the plenum), and the lower tie plate region (the remainder of the fuel assembly below the active fuel region).

tie plate region (the remainder of the fuel assembly below the active fuel region).

9. The flux factors applied to each fuel assembly non-active fuel region are taken from the OCRWM database (DOE 1992). Note that these flux factors are generic and hence applicable to all the fuel designs from all the vendors. It is expected that the actual measured flux data will result in lower flux factors.

Table D-3.1

Fuel Assembly Material Weights for SPC 15x15 PWR Assembly

Material	Weight per Assembly, g
Fuel Zone Active Fuel Region Only	
Zircaloy-4	133927
Inconel-718	112
Uranium	[redacted]
Oxygen	58720
Upper Tie Plate Region	
SS-304	6381
Inconel 718	672
Inconel X-750	24
Plenum Region	
Zircaloy-4	7551
Inconel X-750	3019
Lower Tie Plate Region	
SS-304	4459

The highlighted data is proprietary information to Siemens Power Corporation.

D-3.2 ORIGEN2 Input File Preparation for GE BWR Fuel

The GE BWR fuel assembly is assumed to consist of four axial regions for the source term calculation: the active fuel region, the gas plenum region, the upper tie plate region, and the lower tie plate region.

There are 5 different General Electric (GE) BWR fuel designs from CP&L's Brunswick Nuclear Power Plant that are intended to be shipped in the IF-300 BWR Cask. These fuel designs are identified as GE-7, GE-8, GE-9, GE-10, and GE-13. All of these fuel designs are 8x8 lattices of fuel rods except for GE-13, which is a 9x9 lattice. The GE-7 fuel assemblies contain two small water rods offset diagonally in the center of the lattice. The GE-8 fuel assemblies contain four small water rods in the center of the lattice. Note that two of the four water rods for the GE-8 design are normal water rods, and the other two water rods are empty fuel rods serving as water rods. The GE-9 and GE-10 fuel assemblies have one large water rod located in the center of the lattice. The GE-13 fuel assemblies contain two large water rods offset diagonally in the center of the lattice.

The weights of each material in the four regions were taken from Kunita (1998a) and are listed in Table 3-2.1. Note that Kunita (1998a) does not include the weight of the zircaloy channel in the active fuel and top gas plenum regions. These weights, which are derived below, are added to the weight of the zircaloy in Table 3-2.1 to obtain the total weight of zircaloy in each zone.

A description of the components comprising each of the four zones is given below. The ORIGEN2 analysis conservatively uses the maximum weights for each region from each of the five fuel designs. Table 3-2.2 lists the weights used in the ORIGEN2 analysis. The elemental composition for each of the materials was taken from DOE (1992).

D-3.2.1 Active Fuel Region

The 150 in long active fuel region includes the following components:

- Fuel - UO_2
- Fuel/Water Rod Cladding - Zircaloy-2
- Channel - Zircaloy-2 or 4
- Spacers - Zircaloy-2 or 4
- Spacer Springs- Inconel X-750
- Tabs - Zircaloy-2 or 4

The heavy metal loadings (i.e., UO_2 plus gadolinia rods) for these fuel designs range from [redacted] MTHM (GE-7) to [redacted] MTHM (GE-13) (Kunita 1998). These correspond to uranium loadings of approximately [redacted] MTU (GE-7) and [redacted] MTU (GE-13) when making the conservative assumption that the gadolinia rods are UO_2 rods. The use of the higher fuel loading ([redacted] MTHM) results in higher production of activation products in the non-fuel assembly hardware. The GE-7, 8 and 9 fuel designs have 0.08 in thick channels, and the GE-10 and 13 fuel channels have an effective thickness of 0.07 in. The length and weight of a 0.08 in channel are approximately 166.9 in and 30 kg, respectively, for the Brunswick Nuclear Power Plant which is a GE BWR/4 plant design (DOE 1992). The weight of the channel in the active fuel region is 27 kg (30 kg x 150 in / 166.9 in). It is assumed that the remaining channel material is associated with the top gas plenum region as described below.

D-3.2.2 Upper Tie Plate Region

The upper tie plate region consists of the upper tie plate and the upper locking hardware:

- Upper Tie Plate - 304 stainless steel
- Hex nuts - 304 stainless steel
- Lock Tab Washers - 304 stainless steel

The highlighted data is proprietary information to General Electric.

D-3.2.3 Gas Plenum Region

The plenum region consists of the following components:

- Fuel/Water Rod Cladding - Zircaloy-2
- Channel - Zircaloy-2 or 4
- Plenum Springs- SS & Zircaloy
- Getter - SS & Zircaloy
- End Plugs - Zircaloy-2
- Expansion Springs - Inconel X-750

The weight of the zircaloy channel in the gas plenum region is taken to be 3 kg (30 kg x 16.9 in. / 166.9 in.).

D-3.2.4 Lower Tie Plate Region

The lower tie plate region consists of the following components:

- Bottom Tie Plate - 304 stainless steel
- Finger Springs - Inconel X-750
- End Plugs - Zircaloy-2

D-3.2.5 Modeling Bases and Assumptions

The following lists the modeling bases and assumptions used in the generation of the generation of the source terms for the BWR fuel. Appendix D-9.2 provides input and output files for the ORIGEN2 cases. Section D-5.0 contains a comparison between the results of these ORIGEN2 runs and the current design basis values for the IF-300 cask.

1. The maximum and minimum lattice average enrichment for the GE-7, 8, 9,10 & 13 BWR fuel from Brunswick are 4.25 wt% ^{235}U and 3.19 wt% ^{235}U.

2. The maximum burnup for the BWR fuel is 45 GWD/MTU.

3. The IF-300 Cask with the BWR basket will contain a full load of 17 channeled BWR fuel assemblies.

4. The operating history for the fuel assemblies is assumed to be a high burnup four cycle history similar to the one used in the OCRWM database.

5. A specific power of 25.9 MW/MTU during irradiation is used.

6. The fuel assembly material compositions are taken directly from data supplied by CP&L (Kunita 1998a) and from data contained in DOE (1992).

7. The material weights of burnable poison materials in the higher burnup fuel assembly is not considered in the ORIGEN2 run. This has a very small impact on the overall source strength and decay heat because of their relatively small weights compared to other materials of the fuel assembly.

8. During the irradiation period in the ORIGEN2 model, the fuel assembly is divided into four regions, which are defined as the active fuel region (the portion of the fuel rods associated with the active fuel), the gas plenum region (the portion of the fuel rods above the active fuel length), the upper tie plate region (the remainder of the assembly above the plenum), and the lower tie plate region (the remainder of the fuel assembly below the active fuel region).

9. The flux factors applied to each fuel assembly non-active fuel region are taken from the OCRWM database (DOE 1992). Note that these flux factors are generic and hence applicable to all the fuel designs from all the vendors. It is expected that the actual measured flux data will result in lower flux factors.

Table D-3.2-1

Material Weights (kg) for GE Fuel Designs used in Brunswick (Kunita 1998a)

Region	Material	GE-7	GE-8	GE-9	GE-10	GE-13
Upper Tie Plate	SS-304	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Gas Plenum	SS-304	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Zirc-2,4	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Inconel X-750	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Active Fuel	Zirc-2,4	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Inconel X-750	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Lower Tie Plate	SS-304	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Zirc-2,4	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	Inconel X-750	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

Table D-3.2-2

Fuel Assembly Material Weights used in ORIGEN2 For a GE BWR Assembly[a]

Material	Weight per Assembly, g
Fuel Zone Active Fuel Region Only	
Zircaloy-2	[redacted]
Zircaloy-4	[redacted]
Inconel X-750	[redacted]
Uranium	[redacted]
Oxygen	25000
Upper Tie Plate Region	
SS-304	[redacted]

Material	Weight per Assembly, g
Plenum Region	
Zircaloy-2	[redacted]
Zircaloy-4	[redacted]
Inconel X-750	[redacted]
SS-304	[redacted]
Lower Tie Plate Region	
Zircaloy-2	[redacted]
SS-304	[redacted]
Inconel X-750	[redacted]

[a] The ORIGEN2 analysis conservatively uses the maximum weights for each region from each of the five fuel designs from Table 3-2.1.

The highlighted data is proprietary information to General Electric.

D-3.3 Results for SPC 15X15 PWR Fuel

The ORIGEN2 calculated neutron and gamma sources, decay heat and fission gas generated for the SPC PWR fuel are included in this section. Section D-3.4 contains the results for the GE BWR fuel.

D-3.3.1 Bounding Case for Source Term

By examining the results from the ORIGEN2 model outputs (included in Appendix D-9.1), it is seen that the 50 GWD/MTU burnup with a minimum initial enrichment of 3.45 wt% U-235 produces the maximum neutron and gamma source terms. The results are summarized in Table D-3.3. Therefore, the neutron and gamma sources with 50 GWD/MTU burnup and a minimum initial enrichment of 3.45 wt% U-235 are used as the bounding case for the source term comparison to the current design basis values of the IF-300 cask (CSAR 1995).

D-3.3.2 Neutron Source Calculations

ORIGEN2 includes neutron emissions from (α,n) reactions and spontaneous fission (SF) reactions. The dominant neutron source is SF from ^{244}Cm. The total neutron emission rate from ORIGEN2 for 3.45 wt% ^{235}U PWR fuel with a burnup of 50 GWD/MTU and a decay time of 5 years is 6.80E+08 neutrons/sec/assembly. Therefore, the total neutron source emission rate for the cask loaded with 6 SPC PWR fuel assemblies is:

Total Neutron Emission Rate = 6.80E+08 n/s/FA x 6 FA/cask
= 4.08E+09 n/s/cask

D-3.3.3 Gamma Source Strength Calculation

The gamma particle source strength from the activation products, actinides plus daughters, and fission products in all of the fuel assembly regions are summed to give the total gamma particle source strength for the whole fuel assembly. The total gamma emission rate from ORIGEN2 for the bounding case is 9.23E+15 gamma/sec/assembly. Therefore, the total gamma emission rate for the cask loaded with 6 SPC PWR fuel assemblies is:

Total Gamma Emission Rate = 9.23E+15 γ/s/FA x FA/cask
= 5.54E+16 γ/s/cask

D-3.3.4 Decay Heat Calculation

The total decay heat for the SPC PWR fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years is 1.397 kW (4,768 Btu/h) per assembly. This corresponds to a total cask inner cavity heat load of:

Total Cask Inner Cavity Heat Load = 6 FA/Cask x 4,768 Btu/h/FA
= 28,608 Btu/h.

D-3.3.5 Fission Gas Product Source (^{85}Kr) Calculation

The ^{85}Kr fission gas generated from the fuel assemblies is 4,387 Ci/FA. Note that the higher enriched fuel (i.e., 4.65 wt% ^{235}U) produces more fission gases than the lower enriched fuel (i.e., 3.45 wt% ^{235}U) for a given burnup. Using the same method as the CSAR (1994), the ^{85}Kr inventory available for release from the IF-300 cask is 1,535 Ci/FA. This corresponds to a total cask releasable inventory of ^{85}Kr of:

Total Releasable Inventory of ^{85}Kr = 6 FA/Cask x 1,535 Ci/FA
= 9,210 Ci/cask

D-3.3.6 Fission Gas Product Moles Calculation

The ^{131m}Xe and ^{85}Kr are the two main fission gas products available for release in the cask inner cavity. The volume of ^{131m}Xe at 45 GWD/MTU and 55 GWD/MTU burnups for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor is 206 liters/assembly and 259 liters/assembly at STP respectively (DOE/ET/34014-10 1983). Similarly, the volume of ^{85}Kr at 45 GWD/MTU and 55 GWD/MTU burnups for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor is 21.9 liters/assembly and 27.4 liters/assembly at STP respectively (DOE/ET/34014-10, 1983). This results in values of 24.7 liters/assembly of ^{85}Kr and 233 liters/assembly of ^{131m}Xe at 50 GWD/MTU for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor. These values were obtained by making the assumption that the fission gas product volume varies linearly between 45 and 55 GWD/MTU. The volume of fission gases for the licensing basis burnup of 35 GWD/MTU is 155 liters/assembly of ^{131m}Xe and 16.7 liters/assembly of ^{85}Kr for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor (DOE/ET/34014-10 1983).

Therefore, the volume of the additional fission gas products due to an increase in burnup from 35 GWD/MTU to 50 GWD/MTU is 233-155 = 78 liters/assembly of ^{131m}Xe and 24.7-16.7=8.0 liters/assembly of ^{85}Kr, respectively. For the six PWR fuel assemblies in the cask inner cavity, the total volume of the additional fission gas products due to increase in burnup from 35 GWD/MTU to 50 GWD/MTU is (78 +8.0) liters/assembly * 6 = 516 L (18.22 ft^3). It is expected that no fuel rods will rupture. If in the extreme case it is assumed that all the fuel rods from all six PWR assemblies in the cask inner cavity rupture, then the residual gases in the fuel rods will increase the cask inner cavity pressure.

The volume of the additional fission gas products at the reactor operating conditions is calculated as follows:

$$\frac{P_1V_1}{T_1} = \frac{P_2V_2}{T_2}$$

$$\frac{14.7 \text{ psia} * 18.22 \text{ ft}^3}{(32+460) \text{ R}} = \frac{2500 \text{ psia} * V_2}{(900+460) \text{ R}}$$

which results in V_2 = 0.30 ft^3, where V_2 is the volume of the additional fission gas at the reactor condition of 2500 psia (end-of-life rod pressure) and 900 F (rod gas temperature) (CSAR 1985).

The total fission gas product volume at 50 GWD/MTU available for release is then 1.5 +0.3 = 1.8 ft^3, where 1.5 ft^3 is the total gas volume in all rods available for release at the existing licensed condition of 35 GWD/MTU burnup (CSAR 1985). The number of moles, n, of residual gas that could be released into the cask cavity is estimated to be 0.31 moles which is well below the analysis basis value of 0.5 moles (page 6-50 of CSAR 1985):

$$N = \frac{P_rV_g}{RT_r} = \frac{2500*1.8}{10.73*(900+460)} = 0.31 \text{ moles}$$

Table D-3.3

Summary of Key Evaluation Parameters for SPC 15x15 PWR Fuel With an Exposure of 50 GWD/MTU and a 5 Year Cooling Time

PARAMETER	TOTAL PER ASSEMBLY	TOTAL FOR CASK WITH 6 FAS
NEUTRON EMISSION RATE (NEUTRON/S)	6.80E+08	4.08E+09
GAMMA EMISSION RATE (GAMMA/S)	9.23E+15	5.54E+16
DECAY HEAT (BTU/H)	4,768	28,608
RELEASABLE ^{85}KR FISSION GAS (Ci)	1,536	9,210
TOTAL RESIDUAL RELEASABLE FISSION GAS PRODUCT MOLES	NA	0.31
	(Ci OF ^{60}CO/KG COBALT)	
HARDWARE ACTIVATION	1.33E+04	NA

D-3.4 Results for GE BWR Fuel

The ORIGEN2 calculated neutron and gamma sources, decay heat and fission gas generated for the GE BWR fuel are included in this section.

D-3.4.1 Bounding Case for Source Term

By examining the results from the ORIGEN2 model outputs (included in Appendix D-9.1), it is seen that the 45 GWD/MTU burnup with a minimum initial enrichment of 3.19 wt% U-235 produces the maximum neutron and gamma source terms. The results are summarized in Table D-3.4. Therefore, the neutron and gamma sources with 45 GWD/MTU burnup and a minimum initial enrichment of 3.19 wt% U-235 are used as the bounding case for the source term comparison to the current design basis values of the IF-300 cask (CSAR 1995).

D-3.4.2 Neutron Source Calculations

ORIGEN2 includes neutron emissions from (α,n) reactions and spontaneous fission (SF) reactions. The dominant neutron source is SF from ^{244}Cm. The total neutron emission rate from ORIGEN2 for 3.19 wt% ^{235}U

PWR fuel with a burnup of 45 GWD/MTU and a decay time of 4 years is 2.95E+08 neutrons/sec/assembly. Therefore, the total neutron source emission rate for the cask loaded with 17 GE BWR fuel assemblies is:

Total Neutron Emission Rate = 2.95E+08 n/s/FA x 17 FA/cask
= 5.02E+09 n/s/cask

D-3.4.3 Gamma Source Strength Calculation

The gamma particle source strength from the activation products, actinides plus daughters, and fission products in all of the fuel assembly regions are summed to give the total gamma particle source strength for the whole fuel assembly. The total gamma emission rate from ORIGEN2 for the bounding case is 4.15E+15 gamma/sec/assembly. Therefore, the total gamma emission rate for the cask loaded with 17 GE BWR fuel assemblies is:

Total Gamma Emission Rate = 4.15E+15 γ/s/FA x 17 FA/cask
= 7.06E+16 γ/s/cask

D-3.4.4 Decay Heat Calculation

The total decay heat from the GE BWR fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years is 0.622 kW (2,123 Btu/h) per assembly. This corresponds to a total cask inner cavity heat load of:

Total Cask Inner Cavity Heat Load = 17 FA/Cask x 2,123 Btu/h/FA
= 36,091 Btu/h.

D-3.4.5 Fission Gas Product Source (^{85}Kr) Calculation

The ^{85}Kr fission gas generated from the fuel assemblies is 1,669 Ci/FA. Note that the higher enriched fuel (i.e., 4.25 wt% ^{235}U) produces more fission gases than the lower enriched fuel (i.e., 3.19 wt% ^{235}U) for a given burnup. Using the same method as the CSAR (1994), the ^{85}Kr inventory available for release from the IF-300 cask is 584 Ci/FA. This corresponds to a total cask releasable inventory of ^{85}Kr of:

Total Releasable Inventory of ^{85}Kr = 17 FA/Cask x 584 Ci/FA
= 9,928 Ci/cask

D-3.4.6 Fission Gas Product Moles Calculation

The ^{131m}Xe and ^{85}Kr are the two main fission gas products available for release in the cask inner cavity. The volume of ^{131m}Xe at 40 GWD/MTU and 50 GWD/MTU burnups for an 8X8 GE BWR fuel assembly is 206 liters/assembly and 259 liters/assembly, respectively, at STP (DOE/ET/34014-10 1983). Similarly, the volume of ^{85}Kr at 40 GWD/MTU and 50 GWD/MTU burnups is 21.9 liters/assembly and 27.4 liters/assembly, respectively, at STP (DOE/ET/34014-10 1983). This results in values of 24.7 liters/assembly of ^{85}Kr and 233 liters/assembly of ^{131m}Xe at 45 GWD/MTU for an 8X8 GE BWR fuel assembly. These values were obtained by making the assumption that the fission gas product volume varies linearly between 40 and 50 GWD/MTU.
The volume of fission gases for a burnup of 30 GWD/MTU is approximately 155 liters/assembly of ^{131m}Xe and 16.7 liters/assembly of ^{85}Kr for an 8X8 GE BWR fuel assembly (DOE/ET/34014-10 1983). The volume of fission gases for the licensing basis burnup of 35 GWD/MTU is approximately 181 liters/assembly of ^{131m}Xe and 19.3 liters/assembly of ^{85}Kr for an 8X8 GE BWR fuel assembly. These values were obtained by making the assumption that the fission gas product volume varies linearly between 30 and 40 GWD/MTU (DOE/ET/34014-10 1983).

Therefore, the volume of the additional fission gas products due to an increase in burnup from 35 GWD/MTU to 45 GWD/MTU is 233-181 = 52 liters/assembly of ^{131m}Xe

and 24.7-19.3=5.4 liters/assembly of ^{85}Kr, respectively. For the seventeen BWR fuel assemblies in the cask inner cavity, the total volume of the additional fission gas products due to increase in burnup from 35 GWD/MTU to 50 GWD/MTU is (52 +5.4) liters/assembly * 17 = 976 L (34.46 ft^3). It is expected that no fuel rods will rupture. If in the extreme case it is assumed that all the fuel rods from all seventeen BWR assemblies in the cask inner cavity rupture, then the residual gases in the fuel rods will increase the cask inner cavity pressure.

The volume of the additional fission gas products at the reactor operating conditions is calculated as follows:

$$\frac{P_1V_1}{T_1} = \frac{P_2V_2}{T_2}$$

$$\frac{14.7 \text{ psia} * 34.46 \text{ ft}^3}{(32+460)\text{ R}} = \frac{2500 \text{ psia}*V_2}{(900+460)\text{ R}}$$

which results in $V_2 = 0.56$ ft^3, where V_2 is the volume of the additional fission gas at the reactor condition of 2500 psia (end-of-life rod pressure) and 900 F (rod gas temperature) (CSAR 1985) .

The total fission gas product volume at 45 GWD/MTU available for release is then 1.5 +0.56= 2.06 ft^3, where 1.5 ft^3 is the total gas volume in all rods available for release at the existing licensed condition of 35 GWD/MTU burnup (CSAR 1985). The number of moles, n, of residual gas that could be released into the cask cavity is estimated to be 0.35 moles which is well below the analysis basis value of 0.5 moles (page 6-50 of CSAR 1985):

$$N = \frac{P_rV_g}{RT_r} = \frac{2500*2.06}{10.73*(900+460)} = 0.35 \text{ moles}$$

Note that the report referenced for the fission gas data (DOE/ET/34014-10 1983) does not contain data for 9X9 GE BWR fuel. However, because the data contained in the report is fairly generic, it is assumed to apply to the 9X9 GE BWR fuel. Since there is a large margin to the 0.5 mole analysis basis value, this assumption does not impact the conclusion of this analysis.

Table D-3.4

Summary of Key Evaluation Parameters for GE BWR Fuel With an Exposure of 45 GWD/MTU and a 4 Year Cooling Time

PARAMETER	TOTAL PER ASSEMBLY	TOTAL FOR CASK WITH 17 FAS
NEUTRON EMISSION RATE (NEUTRON/S)	2.95E+08	5.02E+09
GAMMA EMISSION RATE (GAMMA/S)	4.15E+15	7.06E+16
DECAY HEAT (BTU/H)	2,123	36,091
RELEASABLE ^{85}KR FISSION GAS (Ci)	584	9,928
TOTAL RESIDUAL RELEASABLE FISSION GAS PRODUCT MOLES	NA	0.35
	(Ci OF ^{60}CO/KG COBALT)	
HARDWARE ACTIVATION	1.29E+04	NA

D-4.0 DETERMINATION OF ACTIVATION PRODUCT AND CRUD SOURCE TERMS

The generation of the activation product and crud source terms is included in this section. Sections D-4.1 and D-4.2 contain the results for the SPC PWR fuel and the GE BWR fuel, respectively.

D-4.1 SPC PWR Fuel

D-4.1.1 Upper Nozzle Hardware Activation Product Source

Activation analysis is performed to determine the impact of higher burnup on the ^{60}Co content of the upper nozzle of the Robinson Plant's fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years.

An activation analysis based on a unit mass (one kg) of ^{60}Co for a burnup of 35 GWD/MTU at 3 years is contained in Appendix D-9.1, ORIGEN2 run CO265US. Also the activation analysis based on a unit mass (one kg) of ^{60}Co for a burnup of 50 GWD/MTU at 5 years is contained in Appendix D-9.1. ORIGEN2 run CO345UE. A comparison of the results is included in Section D-7.

D-4.1.2 Crud Source Terms from the Package

The increase in the fuel assembly burnup limit from 45 GWD/MTU to 50 GWD/MTU and enrichment limit from 4 wt% ^{235}U to 4.65 wt% ^{235}U will impact the releasable source terms from the IF-300 Package. The releasable source terms include the activation products ("crud") adhering to the exterior surfaces of the fuel assemblies and the fission gas products (mainly ^{85}Kr). The determination of fission gas products is included in Section D-3.3.5. The following contains an evaluation of the crud source term for fuel with an enrichment from 3.45 to 4.65 wt% ^{235}U, a fuel assembly burnup of 50 GWD/MTU, and a minimum 5 year cooling time.

Using the same methodology as the CSAR (1995), the crud activity for a 5 year cooling time is calculated to be 494 curies per assembly, as shown in Table D-4.1. The results show that the majority of the crud activity is due to ^{55}Fe at a 5 year cooling time.

The crud source strengths are given as crud density in more recent data (Sandia 1991). The isotopes ^{63}Ni and ^{55}Fe decay by electron capture rather than gamma emission and they are not included in (Sandia 1991). However, the isotope ^{55}Fe, if taken into the human body, constitutes a substantial radiological hazard. Therefore, to be conservative, ^{63}Ni and ^{55}Fe are included in calculating the crud activity.

The maximum "Worst Case Spot" crud densities ($\mu Ci/cm^2$) for various isotopes in a PWR fuel assembly from Sandia (1991) are used to calculate the crud source term at 5 years, as described below. Table D-4.2 lists the "Worst Case Spot" crud densities from Table I-18 of Sandia (1991).

For a SPC 15x15 PWR fuel rod with an OD of 1.077 cm and a length of 386 cm, the surface area per fuel rod is 1,306 cm^2. For a SPC 15x15 PWR guide tube with an OD of 1.298 cm and a length of 386 cm, the surface area per guide tube is 1,574 cm^2. There are 204 fuel rods and 21 instrument & control rod guide tubes in a SPC 15x15 PWR fuel assembly. The total rod surface area for a 15x15 PWR fuel assembly is 2.99E+05 cm^2 (204x1,306 cm^2 + 21x1,574cm^2). This surface area is used to convert the crud density ($\mu Ci/cm^2$) to crud source per fuel assembly (μCi/FA). Per Sandia (1991), applying the maximum spot density to all rods is conservative. The fuel assembly skeleton is also expected to have a similar crud density. The surface area of the fuel assembly skeleton is very small compared to the surface area of the fuel rods, therefore, it's contribution to the total fuel assembly crud source term can be neglected. The results are summarized in Table D-4.2.

The results from Table D-4.2 show that the total crud source per fuel assembly is 495 curies which is slightly higher than the Table D-4.1 value of 494 curies. The majority of the crud activity is due to ^{55}Fe at a 5 year cooling time. The crud density data is based on fuel assemblies with a burnup range of 9,900 MWD/MTU to 58,800 MWD/MTU (Sandia 1991).

D-4.2 Results for GE BWR Fuel

D-4.2.1 Upper Nozzle Hardware Activation Product Source

Activation analysis is performed to determine the impact of higher burnup of the ^{60}Co content of the upper nozzle of the Brunswick Plant's fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years.

An activation analysis based on a unit mass (one kg) of ^{60}Co for a burnup of 35 GWD/MTU at 3 years is contained in Appendix D-7.3, ORIGEN2 run BWRCO265. Also the activation analysis based on a unit mass (1 kg) of ^{60}Co for a burnup of 45 GWD/MTU at 4 years is contained in Appendix D-7.3. ORIGEN2 run BWRCO319. A comparison of the results is included in Section D-7.

D-4.2.2 Crud Source Terms from the Package

The increase in the fuel assembly burnup limit from 35 GWD/MTU to 45 GWD/MTU and enrichment limit from 4 wt% ^{235}U to 4.25 wt% ^{235}U will impact the releasable source terms from the IF-300 Package. The releasable source terms include the activation products ("crud") adhering to the exterior surfaces of the fuel assemblies and the fission gas products (mainly ^{85}Kr). The determination of fission gas products is included in Section D-3.4.5. The following contains an evaluation of the crud source term for fuel with an enrichment from 3.19 to 4.25 wt% ^{235}U, a fuel assembly burnup of 45 GWD/MTU, and a minimum 4 year cooling time.

The crud source strengths are given as crud density ($\mu Ci/cm^2$) in Sandia (1991). The isotopes ^{63}Ni and ^{55}Fe decay by electron capture rather than gamma emission and they are not included in (Sandia 1991). However, the isotope ^{55}Fe, if taken into the human body, constitutes a substantial radiological hazard. Therefore, to be conservative, ^{63}Ni and ^{55}Fe are included in calculating the crud activity.

Table D-4.3 contains the maximum "Worst Case Spot" crud densities ($\mu Ci/cm^2$) for various isotopes in a BWR fuel assembly from Table I-18 (Sandia 1991). These are used to calculate the crud source term at 4 years cooling time, as described below.

For a GE-13 BWR fuel rod with an OD of [redacted] cm and a length of [redacted] cm, the surface area per fuel rod is [redacted] cm^2. The GE-13 BWR assembly design contains 2 large water rods with an OD of 2.489 cm and a length of [redacted] cm, which results in a surface area per water rod of [redacted] cm^2. There are 74 fuel and 2 water rods in the GE-13 fuel design which results in a total fuel plus water rod surface area of [redacted] cm^2 ([redacted] cm^2) per assembly. The channels for the GE-13 fuel assemblies have an inside width of 13.406 cm, a thickness of 0.165 cm along the sides, and a nominal length of approximately 423.9 cm. The surface area of the inside of the channel is 2.27E+04 cm^2 (4x13.406x423.9) and is 2.33E+04 cm^2 (4x(13.406+2x0.165)x423.9) for the outside of the channel. From Nutech (1990), the approximate area of the upper tie plate is 293 cm^2 and the area of the lower nozzle is approximately 1,330 cm^2. The total surface area of the BWR fuel assembly plus channel is 1.60E+05 cm^2 ([redacted] + [redacted] + 2.27E+04 + 2.33E+04 + 293 + 1.33E+03 cm^2). The GE-13 fuel design was used in this evaluation because it had the highest surface area of the five designs evaluated (i.e., GE-7, GE-8, GE-9, GE-10, and GE-13). Note that 99% of the surface area is from the fuel/water rods and the channel.

The surface area is used to convert the crud density ($\mu Ci/cm^2$) to crud source per fuel assembly (μCi/FA). Per (Sandia 1991) applying the maximum spot density to all rods is conservative. Because crud densities for ^{55}Fe and ^{63}Ni are not included in Sandia (1991), the values for these nuclides were derived from the maximum expected crud radioactivity (μCi/FA) data for PWR fuel assemblies from EPRI (1982). The crud density for ^{63}Ni on the BWR fuel was assumed to be the

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same as that for the PWR fuel. The ^{63}Ni crud density value of 1.24 μCi/cm^2 was calculated using the maximum expected crud density of 3.7E5 μCi/FA (EPRI 1982) and dividing by the SPC PWR surface area of 2.99E+05 cm^2 from the previous section. For ^{55}Fe, the crud density ratio of BWR to PWR fuel for ^{59}Fe is used as both are elemental iron activation products. The ^{59}Fe crud density ratio for BWR to PWR fuel (Sandia 1991) is 87/300 = 0.29. The ^{55}Fe crud density value of 1,650 μCi/cm^2 was calculated using the maximum expected crud density value of 1.7E+09 μCi/FA (EPRI 1982) divided by the SPC PWR surface area of 2.99E+05 cm^2, and finally multiplying by the ^{59}Fe BWR to PWR ratio of 0.29. The results are summarized in Table D-4.3.

The results from Table D-4.3 show that the total crud source per fuel assembly is 217 curies. The majority of the crud activity is due to ^{55}Fe at a 5 year cooling time. The crud density data is based on fuel assemblies with a burnup range of 9,900 MWD/MTU to 58,800 MWD/MTU (Sandia 1991).

Table D-4.1

<u>Releasable Crud Source for SPC PWR Fuel at a 5 Year Cooling Time (CSAR 1995 & EPRI 1982)</u>

Nuclide	Half Life Days	μCi/PWR-FA[1] (T= 0 year)	μCi/PWR-FA (T= 5 year)
Co60	1925.3	4.06E+07	2.10E+07
Co58	70.8	4.06E+08	7.00E+00
Mn54	312.7	1.10E+08	1.92E+06
Fe59	44.6	8.70E+07	4.20E-05
Cr51	27.7	7.00E+07	1.00E-12
Fe55	986.2	1.70E+09	4.71E+08
Ni63	36562.5	3.70E+05	3.57E+05
		Total =	4.94E+08

[1] Taken from EPRI (1982).

Table D-4.2

Releasable Crud Source for SPC PWR Fuel
At a 5 Year Cooling Time (Sandia 1991)

Nuclide	Half Life Days	μCi/cm^2 (T= 0 year)[1]	μCi/PWR-FA (T= 0 year)	μCi/PWR-FA (T= 5 year)
Co58	70.8	1.40E+03	4.19E+08	7.23E+00
Co60	1925.2	1.40E+02	4.19E+07	2.17E+07
Mn54	312.7	3.80E+02	1.14E+08	1.99E+06
Cr51	27.7	3.91E+02	1.17E+08	1.68E-12
Fe59	44.6	3.00E+02	8.98E+07	4.34E-05
Zr95	64.0	3.60E+01	1.08E+07	2.78E-02
Fe55[2]	986.2	NA	1.70E+09	4.71E+08
Ni63[2]	36561.5	NA	3.70E+05	3.57E+05
Total =				4.95E+08

[1] Taken from Sandia (1991).

[2] These isotopes are not included in (Sandia 1991). The values used are from EPRI (1982).

Table D-4.3

BWR Crud Source Term at 4 Year
Cooling Time (Sandia 1991)

Nuclide	Half Life Days	μCi/cm^2 (T= 0 year)[1]	μCi/BWR-FA (T= 0 year)	μCi/BWR-FA (T= 4 year)
Co58	70.8	6.30E+01	1.01E+07	6.21E+00
Co60	1925.2	1.25E+03	2.00E+08	1.18E+08
Mn54	312.7	6.25E+02	1.00E+08	3.93E+06
Cr51	27.7	3.50E+01	5.61E+06	7.49E-10
Fe59	44.6	8.70E+01	1.39E+07	1.96E-03
Zr95	64.0	3.00E+01	4.81E+06	6.48E-01
Zn65	244.4	5.60E+01	8.98E+06	1.43E+05
Fe55[2]	986.2	1.65E+03	2.64E+08	9.45E+07
Ni63[3]	36561.5	1.24E+00	1.98E+05	1.93E+05
Total =				2.17E+08

[1] From Sandia (1991).

[2] The crud activity density ratio of BWR to PWR fuel for ^{59}Fe (Sandia 1991) is used as both are elemental iron activation products.

[3] The crud density for ^{63}Ni on the BWR fuel was assumed to be the same as that for the PWR fuel.

D-5.0 COMPARISON WITH IF-300 PACKAGE C OF C

The key evaluation parameters affected by the increased burnup and enrichment are:

- Decay Heat
- Gamma Shielding
- Neutron Shielding
- Hardware Activation
- Releasable ^{85}Kr Fission Gas Source Term
- Releasable Crud Source Term
- Criticality

Section D-5.1 contains a comparison of the key evaluation parameters for the existing analysis-basis and those for the IF-300 cask loaded with six higher burnup and enrichment SPC PWR fuel assemblies from Robinson. Section D-5.2 contains a comparison of the key evaluation parameters for the existing analysis-basis and those for the IF-300 cask loaded with seventeen higher burnup and enrichment GE BWR fuel assemblies from Brunswick. The criticality safety aspects of the evaluation are contained in Section 6.0.

D-5.1 Results for SPC 15x15 PWR Fuel

D-5.1.1 Decay Heat

The IF-300 shipping cask C of C limit is 40,000 Btu/h, which corresponds to 5,714 Btu/h/FA for the analysis-basis fuel assembly heat loading for the cask loaded with seven assemblies. The maximum decay heat from the SPC 15 x 15 PWR fuel assemblies with a burnup of 50 GWD/MTU, a minimum enrichment of 3.45 wt% ^{235}U, and a cooling time of 5 years is 4,768 Btu/h per assembly, or a total of 28,608 Btu/h for a cask loading of six fuel assemblies. Therefore, the total cask heat loading and the fuel assembly heat loading for SPC PWR fuel with a burnup of 50 GWD/MTU and a cooling time of 5 years are well within the IF-300 shipping cask analysis-basis values. The uranium inventory of the Robinson Plant's 15 x 15 SPC fuel (i.e., 0.437 MTU) is less than the IF-300 shipping cask analysis-basis PWR fuel assembly (i.e., 0.475 MTU) which, when combined with the longer cooling time, accounts for the lower decay heat of the assemblies with a burnup of 50 GWD/MTU and a minimum cooling time of 5 years.

Note that because the individual fuel assembly heat loading for the SPC PWR fuel (4,768 Btu/h/FA) is much less than the analysis-basis fuel assembly heat loading (5,714 Btu/h/FA), the fact that only six assemblies are being loaded versus seven in the analysis-basis thermal evaluation does not change the results of the conclusion even though the distribution of the heat sources is different for a cask loaded with six versus seven fuel assemblies.

In conclusion, the thermal analysis which calculated temperatures and pressures in the IF-300 cask for a 40,000 Btu/h heat load will bound the 50 GWD/MTU burnup case with a minimum cooling time of 5 years.

D-5.1.2 Gamma and Neutron Shielding

Gamma Source Term

The total analysis-basis gamma source is 7.9E+16 gamma/s/cask (CSAR 1995), which corresponds to 1.13E+16 gamma/s/FA for the analysis-basis fuel assembly gamma source term for the cask loaded with seven assemblies. The total gamma emission rate for SPC 15 x 15 PWR fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years is 9.23E+15 gamma/s/FA, which results in a total gamma emission rate of 5.54E+16 gamma/s/cask for the IF-300 cask loaded with six assemblies. Therefore, the individual assembly gamma source term and the total cask gamma source for the IF-300 cask loaded with six SPC 15 x 15 PWR fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years are well within the IF-300 shipping cask analysis-basis values.

Once again, the somewhat higher source terms due to higher burnup of the fuel are offset by the longer cooling time and lower uranium inventory of the SPC 15 x 15 PWR fuel assemblies than the IF-300 shipping cask analysis-basis fuel loading.

Neutron Source Term

The total analysis-basis neutron source is 1.87 E+09 neutrons/s/cask (page A-5-9 of CSAR 1995), which corresponds to 2.67E+08 neutrons/s/FA for the analysis-basis fuel assembly neutron source term for the cask loaded with seven assemblies. The total neutron emission rate for SPC 15 x 15 PWR fuel assemblies with a burnup of 50 GWD/MTU, a minimum enrichment of 3.45 wt% ^{235}U, and a cooling time of 5 years is 6.80E+08 neutrons/sec/FA, which results in a total neutron emission rate of 4.08E+09 neutrons/s/cask for the IF-300 cask loaded with six assemblies. Therefore, the analysis-basis assembly neutron source term is exceeded by a factor of 2.55, and the analysis-basis total cask neutron source term is exceeded by a factor of 2.18 for the IF-300 cask loaded with six SPC 15 x 15 PWR fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years. Therefore it will be conservatively assumed that the neutron dose rates are higher by a maximum of a factor of 2.55 for the higher burnup fuel assemblies.

Impact of Change in Gamma and Neutron Source Terms on Dose Rates

For the neutron and gamma sources, the comparison above is based on the total cask cavity neutron and gamma sources for the 17 element channeled BWR fuel basket (CSAR 1995) since the CSAR does not report the total neutron and gamma sources with dry cask inner cavity PWR fuel shipments.

Even though the above dose rates are for 17 channeled BWR fuel assemblies, they are used here for comparison for the following reasons. Except for the cask top head, the same cask body is used for both the PWR and BWR fuel assemblies. The self-shielding provided by the 17 BWR fuel assemblies is very similar to that for the seven PWR assemblies due to similar fuel assembly material densities. The neutron and gamma source spectrum is also similar.

The dose rates at the cask top nozzle locations (upper tie plate) from (CSAR 1995) are not used directly because of the differences in the geometry at this location. At the cask top head location, the gamma dose from the activation product sources in the upper tie-plate is the controlling dose rate. There are no neutron sources in the cask top head location. The contribution from the neutron sources in the active fuel region of the fuel assembly to the dose rate at 2 meters from the top nozzle surface in the cask radial direction is very small compared to the gamma dose rate. Based on the results in the CSAR (1995), the neutron and gamma dose rates at 2 meters from the top nozzle radial centerline are 0.92 mrem/h and 8.5 mrem/h, respectively. The neutron and gamma dose rates at the cask top head locations for the Robinson Plant's SPC PWR fuel assembly will be lower due to the following.

The top nozzle location for the Robinson Plant's SPC PWR fuel assembly is at least 6 inches from the bottom of the cask upper head due to the presence of basket spacers. Therefore, the depleted uranium shield in the cask radial direction provides radial shielding to all of the upper nozzle gamma sources, thereby reducing the gamma dose rates in the nozzle radial directions.

The results from Section D-3.1 show that the activation product gamma source (i.e., ^{60}Co) in the top nozzle is lower by approximately 6% for fuel with a burnup of 50 GWD/MTU, a minimum initial ^{235}U enrichment of 3.45 wt%, and a minimum cooling time of 5 years.

Therefore, even if the total neutron dose rates are conservatively assumed to increase by as much as 255% due to the higher neutron source term for the fuel with a burnup of 50 GWD/MTU, minimum enrichment of 3.45 wt%, and cooling time of 5 years, the total Normal Conditions of Transfer (NCT) dose rate limit of 10 mrem/h at 2 meters from the package surface will still be satisfied. The results for NCT are summarized in Table D-7.1-2.

Based on the results documented in Table 5.4-5 of the CSAR (1995), the 1 meter dose rate at the radial cask centerline location is the bounding accident condition dose rate. The accident condition gamma and neutron dose rates are 18.1 mrem/h and 163.06 mrem/h, respectively. Therefore, even if the total neutron dose rates are conservatively assumed to increase by as much as 255% due to the higher neutron source term for the fuel with a burnup of 50 GWD/MTU, minimum enrichment of 3.45 wt%, and cooling time of 5 years, the total dose rate limit of 1000 mrem/h at 1 meter from the package surface during accident conditions will still be satisfied. The results for Hypothetical Accident Conditions (HAC) are summarized in Table D-7.1-3.

As summarized in Tables D-7.1-2 and D-7.1-3, these results demonstrate that the total cask dose rates for the SPC PWR fuel from Robinson with a burnup of 50 GWD/MTU, and a cooling time of 5 years satisfy the 10 CFR 71 dose rate limits. The higher burnup of these assemblies, when combined with longer cooling times and lower initial uranium loading, results in all parameters, with the exception of the total neutron source strength, being equal to or less than the currently approved operating envelope of the IF-300 shipping cask. Although the neutron dose rates are higher than the analysis-basis values, the total neutron plus gamma dose rates are below the 10 CFR 71 limits.

Note that the existing C of C contains an additional dose rate limit. The C of C states that the cask content be limited such that under NCT prior to transport, 62 times the neutron dose rate plus 6.3 times the gamma dose rate will not exceed 560 mrem/h at a distance of 6 ft from the side of the cask (CSAR 1995 - pg. A-5-2). Using the neutron and gamma radial dose rates from Table D-7.1-2, 62 times the neutron dose rate of 1.81 mrem/h plus 6.3 times the gamma dose rate of 2.33 mrem/h results in a dose rate of 127 mrem/h, which is well below 560 mrem/h and thus satisfies the C of C criteria.

D-5.1.3 Upper Nozzle Hardware Activation

Activation analysis is performed to determine the impact of higher burnup on the ^{60}Co content of the upper nozzle of the Robinson Plan fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years over that of the analysis-basis fuel assembly.

Any increase in activity causes a corresponding increase in local dose rate at the IF-300 shipping cask lid flange where the shielding is somewhat reduced. An activation analysis (included in Appendix D-9.1) based on unit mass (one kg) of ^{60}Co shows that the Co activity of 1.42E+04 Ci ^{60}Co per initial kg of cobalt for a burnup of 35 GWD/MTU and 3 years cooling time (ORIGEN2 run CO265US) is greater than the ^{60}Co activity of 1.33E+04 Ci ^{60}Co per initial kg of cobalt for a burnup of 50 GWD/TMU and 5 years cooling time (ORIGEN2 run CO345UE). It is concluded that the upper nozzle dose rates from hardware activation for the Robinson Plant's fuel assemblies with burnup of 50 GWD/MTU and a cooling time of 5 years will be lower than the design basis fuel assembly values due to the longer cooling time.

D-5.1.4 Releasable Source Terms from the Package

The comparisons documented in the following subsections show that the amount of radioactive material available for release with fuel assemblies having a burnup of 50 GWD/MTU and a minimum cooling time of 5 years is less than the quantities used to determine the cask containment criteria in the CSAR (1995). Therefore, the cask containment criteria of the CSAR is unaffected by the fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years.

D-5.1.4.1 Releasable ^{85}Kr Fission Gas Source Term

The analysis-basis ^{85}Kr inventory available for release from the IF-300 shipping cask during hypothetical accident conditions in the CSAR (1995) is a total of 13,086 Ci (page 10-11 of CSAR 1995). This corresponds to a ^{85}Kr inventory available for release of 1,869 Ci per PWR fuel assembly. From the ORIGEN2 results for a PWR fuel assembly with a burnup of 50 GWD/MTU, a maximum enrichment of 4.65 wt% ^{235}U, and a cooling time of 5 years, the ^{85}Kr fission gas inventory is 4,388 Ci per assembly. Using the same methodology as the CSAR (1995), the maximum ^{85}Kr inventory available for release from the SPC PWR fuel assemblies with a burnup of 50 GWD/MTU, a maximum enrichment of 4.65 wt% ^{235}U, and a cooling time of 5 years is 1,536 curies per fuel assembly. This results in a total releasable inventory of 9,216 Ci of ^{85}Kr for a cask loaded with six assemblies. This is much less than the releasable inventory used in the CSAR (1995).

Note that the cask containment void volume increases slightly for a cask loading of six assemblies relative to a full loading of seven assemblies, which was used in the analysis-basis containment evaluation. This increase in void volume results in a decrease in the releasable ^{85}Kr activity concentration in the cask, which in turn results in a greater allowable leakage rate. Therefore, the containment requirements from the CSAR (1995) that are associated with the release of ^{85}Kr during hypothetical accident conditions bound the case for six SPC PWR fuel assemblies with a burnup of 50 GWD/MTU, a maximum enrichment of 4.65 wt% ^{235}U, and a minimum cooling time of 5 years.

D-5.1.4.2 Releasable Crud Source Term

The analysis-basis crud activity is 1,074 Ci/FA (page 10-11 of CSAR 1995) based on a two year cooled PWR fuel assembly. This corresponds to a total releasable crud inventory of 7,518 Ci/cask. Using the same methodology as the CSAR (1995), the crud activity at a 5 year cooling time is calculated to be 495 Ci/FA, or 2,970 Ci/cask for a loading of six SPC PWR fuel assemblies. This is much smaller than the releasable inventory used in the CSAR (1995).

As mentioned in section D-5.1.4.1 above, the increased void volume for a cask loading of six fuel assemblies decreases the releasable crud activity concentration in the cask, which in turn results in a greater allowable leakage rate. Therefore, the containment requirements from the CSAR (1995) that are associated with the release of crud during hypothetical accident conditions bound the case for six SPC PWR fuel assemblies with a burnup of 50 GWD/MTU, a maximum enrichment of 4.65 wt% ^{235}U, and a minimum cooling time of 5 years.

D-5.2 Results for GE BWR Fuel

D-5.2.1 Decay Heat

The IF-300 shipping cask C of C limit is 40,000 Btu/h, which corresponds to 2,353 Btu/h/FA for the analysis-basis fuel assembly heat loading for the cask loaded with seventeen assemblies. The maximum decay heat from the GE BWR fuel assemblies with a burnup of 45 GWD/MTU, a minimum enrichment of 3.19 wt% ^{235}U, and a cooling time of 4 years is 2,123 Btu/h per assembly, or a total of 36,091 Btu/h for a cask loading of seventeen fuel assemblies. Therefore, the total cask heat loading and the fuel assembly heat loading for GE BWR fuel with a burnup of 45 GWD/MTU and a cooling time of 4 years are well within the IF-300 shipping cask analysis-basis values. The uranium inventory of the Brunswick Plant's fuel (i.e., 0.186 MTU) is less than the IF-300 shipping cask analysis-basis BWR fuel assembly (i.e., 0.198 MTU) which, when combined with the longer cooling time, accounts for the lower decay heat of the assemblies with a burnup of 45 GWD/MTU and a minimum cooling time of 4 years.

In conclusion, the thermal analysis which calculated temperatures and pressures in the IF-300 cask for a 40,000 Btu/h heat load will bound the 45 GWD/MTU burnup case with a minimum cooling time of 4 years.

D-5.2.2 Gamma and Neutron Shielding

Gamma Source Term

The total analysis-basis gamma source is 7.9E+16 gamma/s/cask (CSAR 1995), which corresponds to 4.65E+15 gamma/s/FA for the analysis-basis fuel assembly gamma source term for the cask loaded with seventeen assemblies. The total gamma emission rate BWR fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years is 4.15E+15 gamma/s/FA, which results in a total gamma emission rate of 7.06E+16 gamma/s/cask for the IF-300 cask loaded with seventeen assemblies. Therefore, the individual assembly gamma source term and the total cask gamma source for the IF-300 cask loaded with seventeen channeled BWR fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years are well within the IF-300 shipping cask analysis-basis values.

Once again, the somewhat higher source terms due to higher burnup of the fuel are offset by the longer cooling time and lower uranium inventory of the BWR fuel assemblies than the IF-300 shipping cask analysis-basis fuel loading.

Neutron Source Term

The total analysis-basis neutron source is 1.87 E+09 neutrons/s/cask (page A-5-9 of CSAR 1995), which corresponds to 1.10E+08 neutrons/s/FA for the analysis-basis fuel assembly neutron source term for the cask loaded with seven assemblies. The total neutron emission rate for BWR fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years is 2.95E+08 neutrons/sec/FA, which results in a total neutron emission rate of 5.02E+09 neutrons/s/cask for the IF-300 cask loaded with seventeen assemblies. Therefore, the analysis-basis assembly and total cask neutron source terms are exceeded by a factor of 2.68 for the IF-300 cask loaded with seventeen BWR fuel assemblies with a burnup of 45 GWD/MTU, a minimum enrichment of 3.19 wt% ^{235}U, and a cooling time of 4 years. Therefore it will be conservatively assumed that the neutron dose rates are higher by a maximum of a factor of 2.68 for the higher burnup fuel assemblies.

Impact of Change in Gamma and Neutron Source Terms on Dose Rates

For the neutron and gamma sources, the comparison above is based on the total cask cavity neutron and gamma sources for the 17 element channeled BWR fuel basket (CSAR 1995).

The gamma dose from the activation product sources in the upper tie-plate at the cask top head location is the controlling dose rate. There are no neutron sources in the cask top head location. The contribution of the neutron sources from the active fuel region of the fuel assembly to the dose rate at 2 meters from the top nozzle surface in the cask radial direction is very small compared to the gamma dose rate. Based on the results in the CSAR (1995), the neutron and gamma dose rates at 2 meters from the top nozzle radial centerline are 0.92 mrem/h and 8.5 mrem/h, respectively. The neutron and gamma dose rates at the cask top head locations for the GE BWR fuel will be lower because the activation product gamma source in the top nozzle is lower by approximately 1.5% for fuel with a burnup of 45 GWD/MTU, a minimum initial ^{235}U enrichment of 3.19 wt%, and a minimum cooling time of 4 years (see Section 3.2).

Therefore, even if the total neutron dose rates are conservatively assumed to increase by as much as 268% due to the higher neutron source term for the fuel with a burnup of 45 GWD/MTU, minimum enrichment of 3.19 wt%, and cooling time of 4 years, the total dose rate limit of 10 mrem/h at 2 meters from the package surface will still be satisfied. The results are summarized in Table D-7.2-2.

Based on the results documented in Table 5.4-5 of the CSAR (1995), the 1 meter dose rate at the radial cask centerline location is the bounding accident condition dose rate. The accident condition gamma and neutron dose rates are 18.1 mrem/h and 163.06 mrem/h, respectively. Therefore, even if the total neutron dose rates increase by as much as 268% due to the higher neutron source term for the fuel with a burnup of 45 GWD/MTU, minimum enrichment of 3.19 wt%, and cooling time of 4 years, the total dose rate limit of 1000 mrem/h at 1 meter from the package surface during accident conditions will still be satisfied. The results are summarized in Table D-7.2-3.

As summarized in Table D-7.2-1, these results demonstrate that the total cask dose rates for the BWR fuel from Robinson with a burnup of 45 GWD/MTU, and a cooling time of 4 years satisfy the 10 CFR 71 dose rate limits. The higher burnup of these assemblies, when combined with longer cooling times and lower initial uranium loading, results in all parameters, with the exception of the total neutron source strength, being equal to or less than the currently approved operating envelope of the IF-300 shipping cask. Although the neutron dose rates are higher than the analysis-basis values, the total neutron plus gamma dose rates are below the 10 CFR 71 limits.

Note that the existing C of C contains an additional dose rate limit. The C of C states that the cask content be limited such that under NCT prior to transport, 62 times the neutron dose rate plus 6.3 times the gamma dose rate will not exceed 560 mrem/h at a distance of 6 ft from the side of the cask (CSAR 1995 - pg. A-5-2). Using the neutron and gamma radial dose rates from Table D-7.2-2, 62 times the neutron dose rate of 1.90 mrem/h plus 6.3 times the gamma dose rate of 2.33 mrem/h results in a dose rate of 132 mrem/h, which is well below 560 mrem/h and thus satisfies the C of C criteria.

D-5.2.3 Upper Nozzle Hardware Activation

Activation analysis is performed to determine the impact of higher burnup on the ^{60}Co content of the upper nozzle of the Brunswick Plant's fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years over that of the analysis-basis fuel assembly. Any increase in activity causes a corresponding increase in local dose rate at the IF-300 shipping cask lid flange where the shielding is somewhat reduced. An activation analysis (included in Appendix D-9.2) based on unit mass (one kg) of ^{60}Co shows that the Co activity of 1.31E+04 Ci ^{60}Co per initial kg of cobalt for a burnup of 35 GWD/MTU and 3 years cooling time (ORIGEN2 run BWRCO265) is greater than the ^{60}Co activity of 1.29E+04 Ci ^{60}Co per initial kg of cobalt for a burnup of 45 GWD/TMU and 4 years cooling time (ORIGEN2 run BWRCO319). It is concluded that the upper nozzle dose rates from hardware activation for the Brunswick Plant's fuel assemblies with burnup of 45 GWD/MTU and a cooling time of 4 years will be lower than the design basis fuel assembly values due to the longer cooling time.

D-5.2.4 Releasable Source Terms from the Package

The analysis-basis ^{85}Kr inventory available for release from the IF-300 shipping cask during hypothetical accident conditions in the CSAR (1995) is a total of 13,086 Ci (page 10-11 of CSAR 1995). This corresponds to a ^{85}Kr inventory available for release of 770 Ci per BWR fuel assembly. From the ORIGEN2 results for a BWR fuel assembly with a burnup of 45 GWD/MTU and a cooling time of 4 years, the ^{85}Kr fission gas inventory is 1,666 Ci per assembly. Using the same methodology as the CSAR (1995), the maximum ^{85}Kr inventory available for release from the BWR fuel assemblies with a burnup of 45 GWD/MTU, a maximum enrichment of 4.65 wt% ^{235}U, and a cooling time of 4 years is 584 curies per fuel assembly. This results in a total releasable inventory of 9,928 Ci of ^{85}Kr for a cask loaded with seventeen assemblies. This is much less than the releasable inventory used in the CSAR (1995).

Therefore, the containment requirements from the CSAR (1995) that are associated with the release of ^{85}Kr during hypothetical accident conditions bound the case for BWR fuel with a burnup of 45 GWD/MTU, a maximum enrichment of 4.65 wt% ^{235}U, and a minimum cooling time of 4 years.

D-5.2.4.2 <u>Releasable Crud Source Term</u>

The analysis-basis crud activity is 1,074 Ci/FA (page 10-11 of CSAR 1995) based on a two year cooled PWR fuel assembly. This corresponds to a total releasable crud inventory of 7,518 Ci/cask. Using the same methodology as the CSAR (1995), the crud activity at a 4 year cooling time is calculated to be 217 Ci/FA, or 3,689 Ci/cask for a loading of seventeen BWR fuel assemblies. This is much smaller than the releasable inventory used in the CSAR (1995).

Therefore, the containment requirements from the CSAR (1995) that are associated with the release of crud during hypothetical accident conditions bound the case for BWR fuel with a burnup of 45 GWD/MTU, a maximum enrichment of 4.25 wt% ^{235}U, and a minimum cooling time of 4 years.

D-6.0 CRITICALITY EVALUATION

This chapter describes the engineering/physics design elements of the IF-300 Cask which are important to safety and necessary to comply with the performance requirements specified in Sections 71.55 and 71.59 of 10 CFR Part 71.

The results of detailed analyses are presented which demonstrate that the IF-300 Cask is critically safe under normal conditions of transfer (NCT) and hypothetical accident conditions (HAC), considering a variety of uncertainties. Sufficient detail has been included to permit an independent evaluation of the criticality analyses. Much of this detail is available in Section D-6.9 in which computer inputs have been provided for review.

D-6.1 Description of Criticality Design

D-6.1.1 Packaging Design Features

The IF-300 Shipping Cask baskets are designed to provide criticality control through a combination of mechanical and neutronic isolation of fuel assemblies. A support structure composed of four axially oriented support rods and nine spacer disks provides positive location for the fuel assemblies under both normal and accident conditions. The basket assembly uses fixed neutron absorbers which effectively isolate pairs of fuel assemblies.

The IF-300 Shipping Cask has three different baskets to accommodate Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fuel. The PWR Cask basket has a capacity of seven PWR fuel assemblies and uses rods containing B_4C for criticality control. There are currently two BWR basket designs licensed for use in the IF-300 cask. The original licensed BWR basket design is capable of transporting eighteen unchanneled BWR fuel assemblies, and uses rods containing B_4C for criticality control. The later basket design allows for loading seventeen channeled BWR fuel assemblies, and uses borated stainless steel plates for criticality control. This evaluation focuses on the original licensed PWR basket and the later BWR basket design which is capable of transporting 17 channeled BWR fuel assemblies. The licensed contents for the earlier BWR basket design capable of transporting 18 unchanneled BWR fuel assemblies is not affected by this evaluation.

The PWR basket contains rods containing B_4C for criticality control. These rods are 0.5 in diameter (0.02 in wall) 304 Stainless Steel (SS) tubes filled with B_4C having an average compacted density of 1.76 ± 0.13 g/cc of natural boron with 18.3 ± 0.3 wt% ^{10}B (Vectra 1995). The B_4C rods are positioned between the axial spacer disks, and vary in length from 8.75 in to 21.5 in. These absorber rods are manufactured by the General Electric Company following the same standards, where applicable, used for BWR control blade absorber tubes. Quality control checks include B_4C density determinations, helium leak checking and material certifications on both tubing and end plugs.

The BWR basket contains borated stainless steel plates for criticality control. The borated plates are made of Neutrosorb Plus™, which is a modified Type 304 austenitic chrome-nickel stainless steel which can be supplied with boron additions up to 2% to provide thermal neutron absorption. A standards specification for borated stainless steels denoted ASTM A887-88 (ASTM 1989) has been approved by the American Society for Testing and Materials (ASTM). Neutrosorb Plus™ is manufactured to the ASTM A887 specification using powder metallurgy techniques which have been shown to exhibit better mechanical properties and homogeneity than conventional cast/wrought products, B_4C aggregate products, or coatings. The absorber material is isotopically enriched thus allowing the plates to have a lower boron content which significantly improves its mechanical properties relative to borated stainless steel with natural boron.

The material properties for these plates are as follows (CSAR 1995):

Finished plate thickness	0.250" to 0.375" +0.045"/-0.010"
SS304 "Skin Thickness"	0.007" maximum, each side
Boron Content	1.0 w/o minimum
B-10 Enrichment	94% (by atoms) B-10, min
Plate density	7.76 g/cc minimum

The principal performance features from 10 CFR 71 for the IF-300 cask as they relate to criticality control are shown in Table D-6.1-1.

Table D-6.1-1

Performance Requirements (10 CFR 71.55 and 10 CFR 71.59)

General Requirements		
	A package must be subcritical if water were to leak into the containment system under the following conditions:	
	1	Most reactive credible configuration consistent with the chemical and physical form of the material.
	2	Moderation by water to the most reactive credible extent.
	3	Close full reflection of the containment system by water on all sides or such greater reflection of the system as may be provided by the material of the packaging.
Normal Conditions of Transport		
	1	The contents would be subcritical.
	2	The geometric form of the package contents would not be substantially altered.
	3	There would be no leakage of water into the containment system unless in the evaluation of undamaged packages it is assumed that moderation is present to cause maximum reactivity.
	4	No substantial reduction in the effectiveness of the packaging: • No more than 5% reduction in total volume • No more than 5% reduction in effective spacing • No occurrence of an aperture in the outer surface of the packaging large enough to admit a 4-inch cube.
Hypothetical Accident Conditions		
	The package is subcritical under the following conditions:	
	1	Most reactive credible configuration consistent with the chemical and physical form of the contents.
	2	Moderation by water to the most reactive credible extent consistent with the chemical and physical form of the contents.
	3	Full reflection by water on all sides.
Standards for Arrays		
	A package must be controlled to assure than an array remains subcritical. Derive a number N such that assuming packages are stacked together in any arrangement and with close full reflection on all sides of the stack by water:	
	1	5 times N undamaged packages with nothing between the packages would be subcritical
	2	2 times N damaged packages would be subcritical with optimum interspersed hydrogenous moderation.
	3	$N \geq 0.5$
Transport Index (TI)		
	Divide 50 by N: TI=100 if N=0.5 .	

D-6.1.2 Summary of Criticality Evaluation Results

Table D-6.1-2 contains a summary of the key criticality evaluation results for the IF-300 Cask.

Table D-6.1-2

Summary Table of Criticality Evaluations

	6 PWR Fuel Assemblies with 4.65 wt% ^{235}U	17 BWR Fuel Assemblies with 4.25 wt% ^{235}U
Normal Conditions of Transport (NCT)		
Number of Undamaged Packages	∞	∞
Single Unit Maximum k_{tot}[a]	0.34916	0.33984
Infinite Array Maximum k_{tot}[a]	0.38870	0.37240
k_{eff} Limit	0.95	0.95
Hypothetical Accident Conditions (HAC)		
Single Unit Maximum k_{tot}[a]	0.94689	0.93185
Infinite Array Maximum k_{tot}[a]	0.94830	0.93407
k_{eff} Limit	0.95	0.95

[a] k_{tot} = k_{eff} including bias and uncertainties (see Sections D-6.4-3 and D-6.7).

D-6.1.3 Transport Index

The maximum calculated k_{tot} is less than 0.95 for infinite arrays of IF-300 casks under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) when the casks are loaded with six 4.65 wt% ^{235}U Siemens Power Corporation (SPC) PWR fuel assemblies with the center basket location empty, or with seventeen 4.25 wt% ^{235}U BWR fuel assemblies. Therefore, a criticality Transport Index of zero (0) is justified.

D-6.2 Spent Nuclear Fuel Contents

Prior to 1991, the IF-300 Cask was licensed for transport of up to seven PWR or eighteen unchanneled BWR irradiated fuel assemblies. The enrichment limit was 4 wt% ^{235}U for both the PWR and BWR fuel. The IF-300 Channeled BWR fuel basket amendment (CSAR 1995) was made to allow shipment of seventeen channeled BWR fuel assemblies in the IF-300 cask with enrichments up to 4 wt% ^{235}U.

This criticality evaluation addresses shipment of higher enriched PWR and BWR fuel assemblies from Carolina Power and Light's Robinson and Brunswick Nuclear Power Plants, respectively. The maximum enrichment for the PWR assemblies is 4.65 wt% ^{235}U, and the maximum enrichment for the BWR assemblies is 4.25 wt% ^{235}U. Since no credit for burnup or burnable poisons is assumed in the criticality calculations, both irradiated and unirradiated fuel is qualified for shipment.

The maximum fuel loading parameters as they relate to criticality are summarized in Table D-6.2-1. Sections 6.2.1 and 6.2.2 provide a description of the allowable PWR and BWR fuel types, respectively .

Table D-6.2-1

Maximum Fuel Loading Parameters

Parameter	PWR Fuel	BWR Fuel
Fuel Type	15x15 SPC PWR for Westinghouse Class 15x15 Reactors	GE-7, 8, 9, 10 &13 BWR Fuel for General Electric BWR/4 Plant Design
Uranium Weight	437 kg/assembly	187 kg/assembly
Number of Assemblies	6[a]	17 channeled
Minimum Assembly Average Burnup	0 GWD/MTU	0 GWD/MTU
Maximum Lattice Average Enrichment	4.65 wt% ^{235}U	4.25 wt% ^{235}U

[a] The center location in the PWR basket must be left empty. If an assembly is loaded in the center PWR basket location and a peripheral location is instead left empty, the cask will not meet the criticality safety criteria with 6 assemblies at 4.65 wt% ^{235}U.

D-6.2.1 SPC 15x15 PWR Fuel Design Description

The Siemens Power Corporation (SPC) 15x15 fuel assembly design is the only PWR fuel from CP&L's Robinson Nuclear Power Plant that is intended to be shipped in the IF-300 cask. This fuel design incorporates GdO_2 integral burnable absorbers and 6 inch top and bottom axial blankets containing natural UO_2 pellets (0.72 wt% ^{235}U) with a [redacted]% theoretical fuel density. The SPC assembly contains 204 fuel rods, 20 guide tubes, and 1 instrument tube. The geometry of a typical fuel rod cell is shown in Figure D-6.2-1. Figures D-6.2-2a and D-6.2-2b illustrate the SPC PWR fuel assembly models used in the evaluation, and Table D-6.2-2 summarizes the fuel design data.

Although the pin-by-pin ^{235}U enrichment varies throughout the fuel lattice in the central enriched zone, all fuel pins in this zone were assumed to contain UO_2 pellets enriched to 4.65 wt% ^{235}U. Although the PWR basket design is capable of holding seven assemblies, this evaluation addresses a cask loading of six assemblies with the center location left empty. This restriction allows the shipment of the higher enrichment (4.65 wt%^{235}U) fuel without requiring any cask or basket design modifications.

Consistent with the requirements contained in NUREG/CR- 5661 (NRC 1997), no credit is taken for fuel burnup or integral burnable absorbers such as gadolinium.

D-6.2.2 GE BWR Fuel Design Description

There are 5 different General Electric (GE) BWR fuel designs from CP&L's Brunswick Nuclear Power Plant that are intended to be shipped in the IF-300 BWR Cask. These fuel designs are identified as GE-7, GE-8, GE-9, GE-10, and GE-13. All of these fuel designs have 8x8 lattices of fuel rods except for GE-13, which has a

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9x9 lattice. Table D-6.2-3 summarizes the fuel-related data for each fuel design.

The GE-7 fuel assemblies contain two small water rods offset diagonally in the center of the lattice (Figure D-6.2-3). The GE-8 fuel assemblies contain four small water rods in the center of the lattice (Figure D-6.2-4). Note that two of the four water rods for the GE-8 design are normal water rods, and the other two water rods are empty fuel rods serving as water rods. The GE-9 and GE-10 fuel assemblies have one large water rod located in the center of the lattice (Figure D-6.2-5). The GE-13 fuel assemblies contain two large water rods offset diagonally in the center of the lattice (Figure D-6.2-6). Figure D-6.2-7 is an axial view of a GE-8 assembly, which is representative of all five of the GE fuel designs of interest.

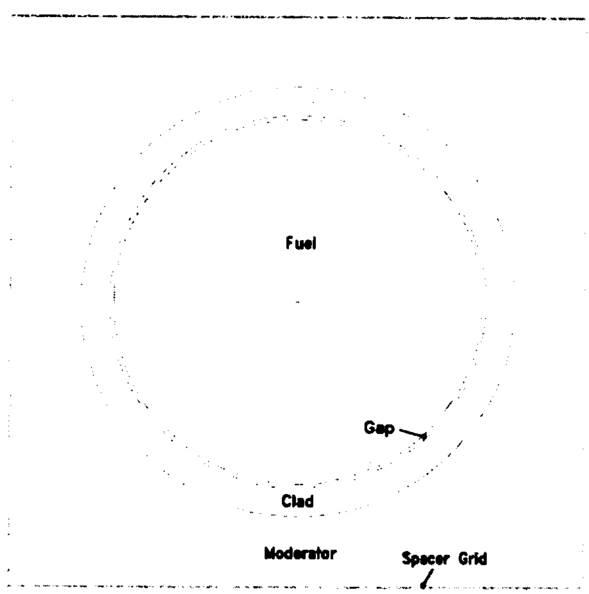
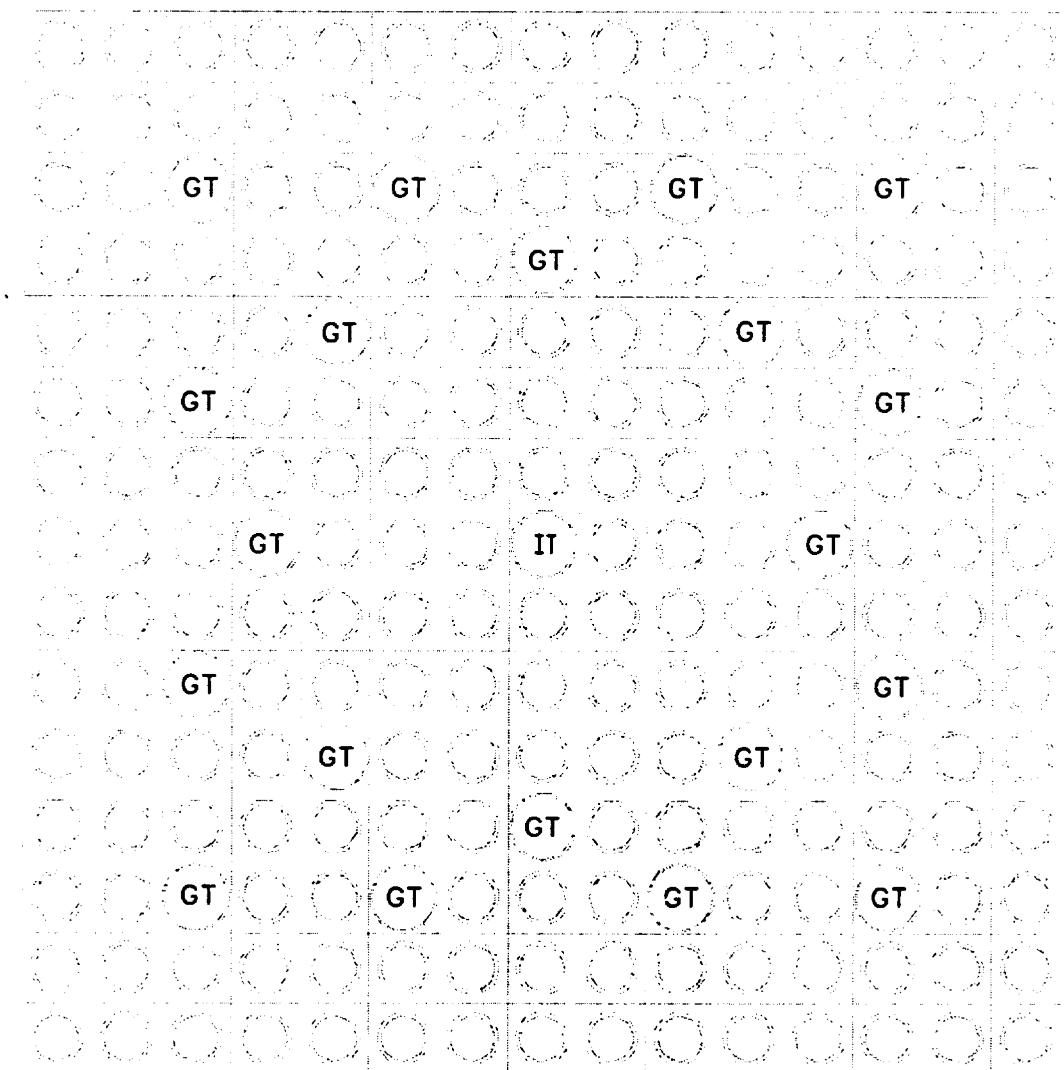


Figure D-6.2-1

<u>Typical Fuel Pin</u>



GT = control rod guide tube.
IT = instrument guide tube.

Figure D-6.2-2a

Depiction of SPC 15x15 Fuel Assembly - Radial View

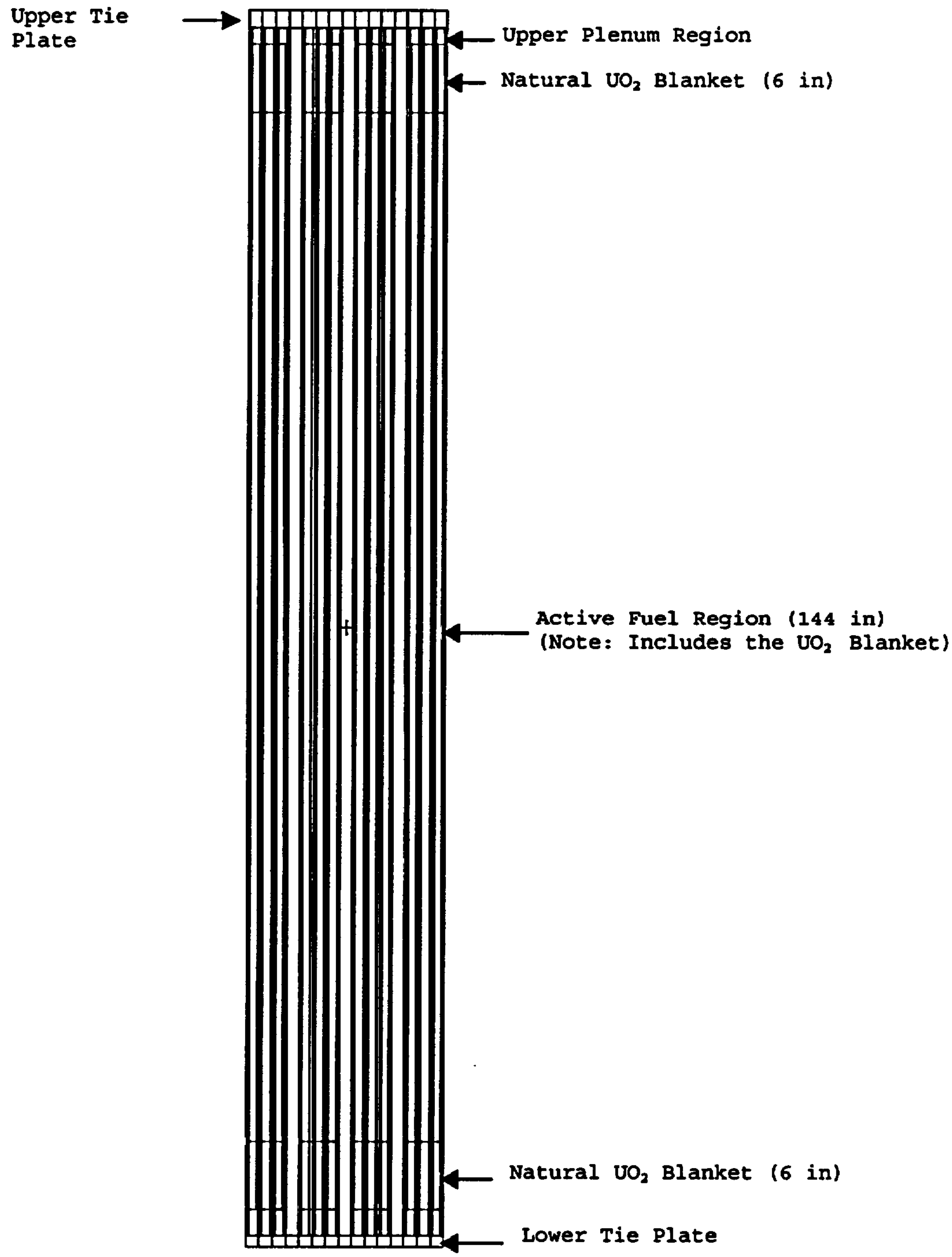


Figure D-6.2-2b

Depiction of SPC 15x15 Fuel Assembly - Axial View

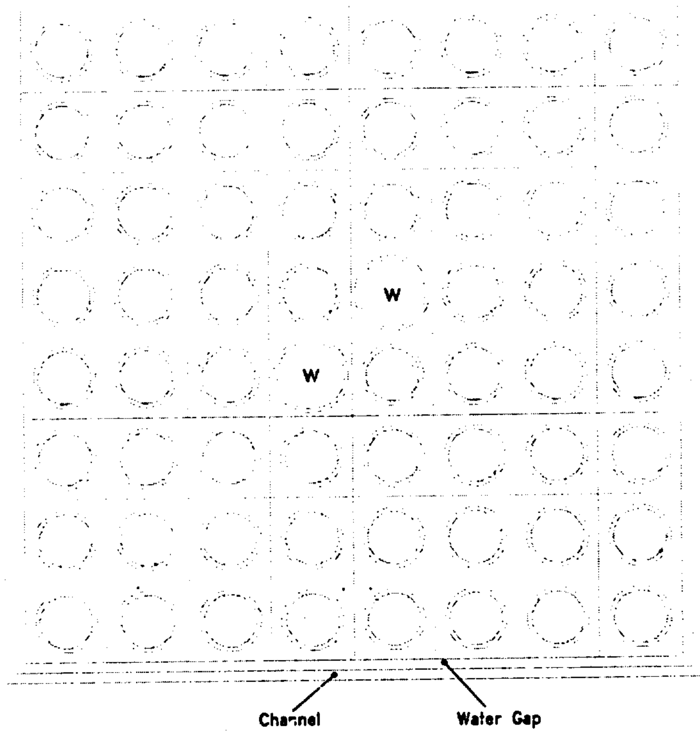


Figure D-6.2-3

Depiction of GE-7 Fuel Assembly

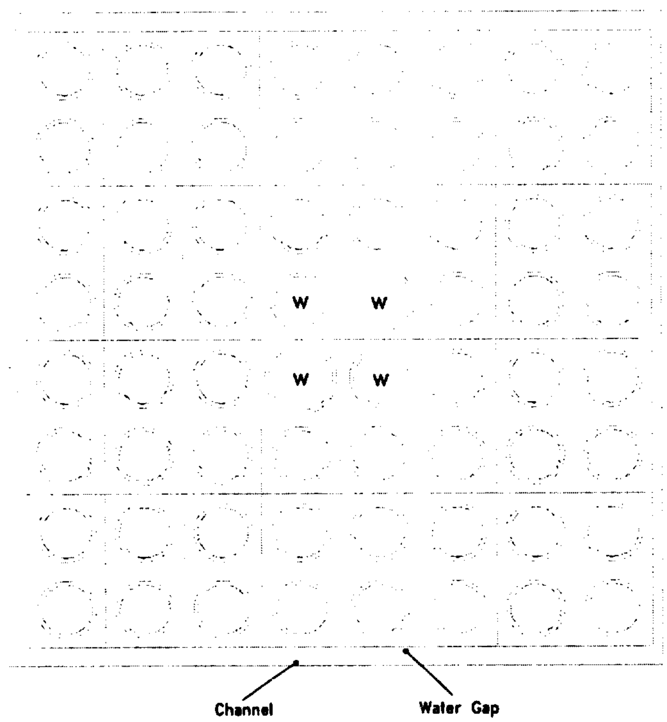


Figure D-6.2-4

Depiction of GE-8 Fuel Assembly

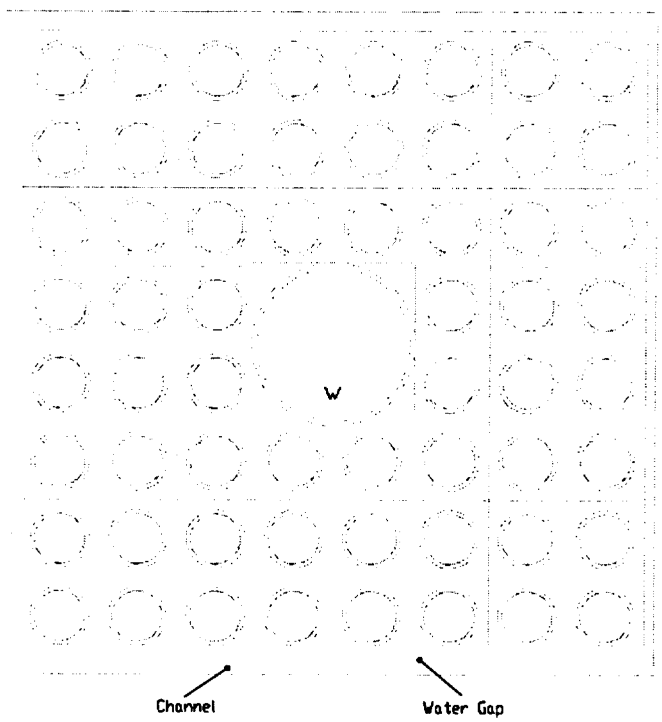


Figure D-6.2-5

Depiction of GE-9 and GE-10 Fuel Assembly

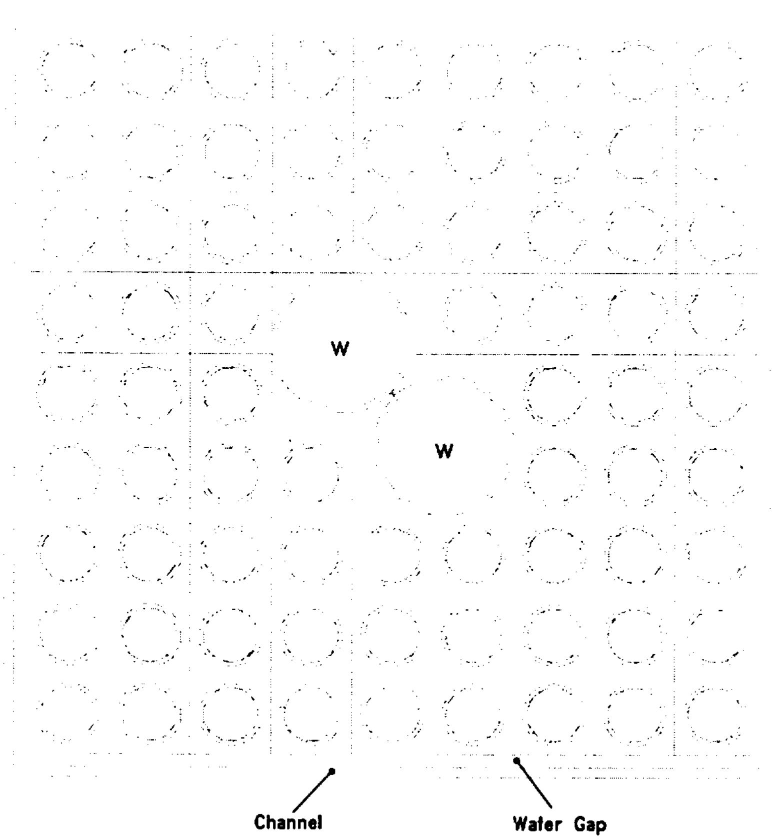


Figure D-6.2-6

Depiction of GE-13 Fuel Assembly

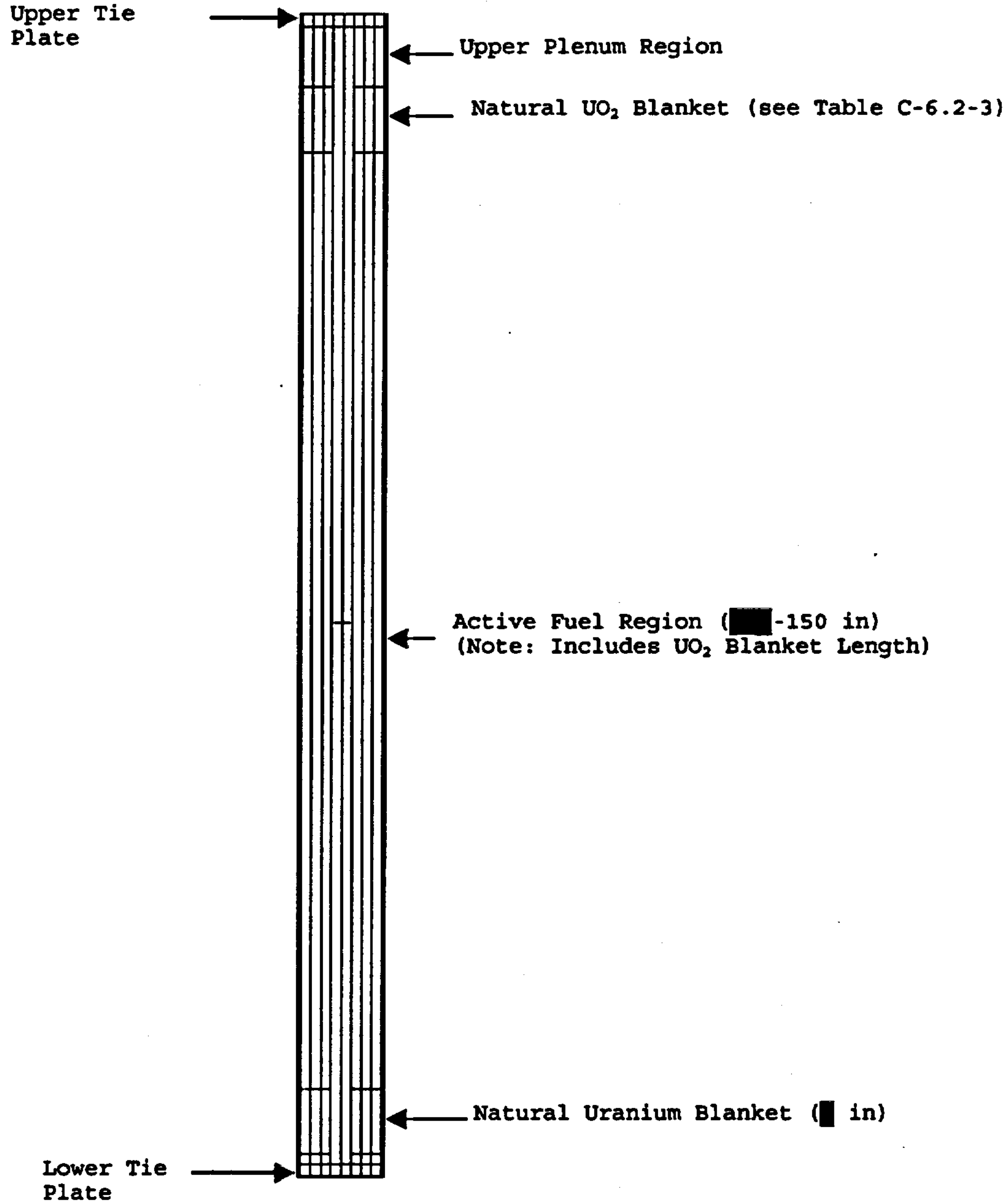


Figure D-6.2-7

Depiction of a GE Fuel Assembly - Axial View

The highlighted data is proprietary information to General Electric.

Table D-6.2-2

SPC 15x15 PWR Fuel Design Summary

ROD ARRAY	15x15
PELLET OD (in)	0.357
CLAD ID (in)	0.364
CLAD OD (in)	0.424
CLAD MATERIAL	Zr-4
FUEL ROD PITCH (in)	0.563
UPPER GUIDE TUBE ID (in)	0.511
UPPER GUIDE TUBE OD (in)	0.544
INSTRUMENT TUBE OD (in)	0.511
INSTRUMENT TUBE ID (in)	0.544
INSTRUMENT/GUIDE TUBE MATERIAL	Zr-4
ACTIVE FUEL HEIGHT (in)	144
FUEL ROD LENGTH (in)	[redacted]
PLENUM LENGTH (in)	[redacted]
ASSEMBLY LENGTH (in)	[redacted]
LTP WEIGHT (kg)	4.117
UTP WEIGHT (kg)	6.381
SPACER GRID WEIGHT (kg)	7.78
SPACER GRID MATERIAL	Zr-4
TOP BLANKET (in)	6
BOTTOM BLANKET (in)	6
PELLET %TD	[redacted]

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Table D-6.2-3

Fuel Design Summary for GE Fuel used at Brunswick

FUEL TYPE	GE-7	GE-8	GE-9	GE-10	GE-13
ROD ARRAY	8 x 8	8 x 8	8 x 8	8 x 8	9 x 9
PELLET OD (in)	0.41	0.411	0.411	0.411	[redacted]
CLAD ID (in)	0.419	0.419	0.419	0.419	[redacted]
CLAD OD (in)	0.483	0.483	0.483	0.483	0.44
CLAD MATERIAL	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
FUEL ROD PITCH (in)	0.64	0.64	0.64	0.64	0.566
WATER ROD ID (in)	0.531	0.531	1.26	1.26	[redacted]
WATER ROD OD (in)	0.591	0.591	1.34	1.34	0.98
WATER ROD MATERIAL	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
ACTIVE FUEL HEIGHT	150	150	150	150	[redacted]
CHANNEL INNER DIM	5.278	5.278	5.278	5.278	5.278
CHANNEL THICKNESS	0.08	0.08	0.08	0.07[a]	0.07[a]
CHANNEL MATERIAL	Zr-4	Zr-4	Zr-2 or	Zr-2	Zr-2
ROD LENGTH (in)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
PLENUM LENGTH (in)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
ASSEMBLY LENGTH	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
LTP WEIGHT (kg)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
UTP WEIGHT (kg)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
SPACER GRID WEIGHT	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
SPACER GRID	Zr-2 or	Zr-2 or	Zr-2	Zr-2	Zr-2
TOP BLANKET (in)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
BOTTOM BLANKET (in)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
PELLET %TD	94.4-98	95.0-98	94.4-98	94.4-98	[redacted]

[b] The channels used for the GE-10 and GE-13 fuel designs are [redacted] inches thick in the corners and [redacted] inches thick along the sides. The effective thickness of the channel is 0.07 inches.

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D-6.3 General Considerations for Evaluations

10 CFR 71 states that the effective neutron multiplication factor (k_{eff}) must not exceed 0.95 after consideration of appropriate bias and uncertainties for the following configurations:

1) a single package with optimum moderation within the containment system, close water reflection, and the most reactive packaging and content configuration (consistent with the effects of normal conditions of transport [NCT] or hypothetical accident conditions [HAC], whichever is more reactive).

2) an array of 5N undamaged packages (packages subject to normal conditions of transport) with nothing between the packages and close water reflection of the array.

3) an array of 2N damaged packages (packages subject to hypothetical accident conditions) if each package were subjected to the tests specified in 10 CFR 71.73, with optimum interspersed moderation and close water reflection of the array.

Normally, the array calculations begin with an infinite array model because, if the infinite array is adequately subcritical under NCT and HAC, no additional array calculations are necessary. If the infinite array is not shown to be safely subcritical, a finite array of packages is analyzed until an array size is found that is adequately subcritical.

The evaluation for the IF-300 cask containing six SPC PWR fuel assemblies with an enrichment of 4.65 wt% ^{235}U, or seventeen channeled BWR fuel assemblies with an enrichment of 4.25 wt% ^{235}U, indicates that an infinite array of packages is safely subcritical. Therefore, all array calculations were done using an infinite array of packages.

D-6.3.1 Computer Codes and Cross Section Libraries

The MCNP Version 4B2 (Breismeister, J. F., 1997) computer code was used for all criticality calculations performed in this evaluation. The MCNP computer code is a general-purpose Monte Carlo code with powerful geometry routines. It uses Evaluated Nuclear Data Files (ENDF/B) for cross sections [BNL 1991]. The state-of-the-art ENDF/B system is maintained by the National Nuclear Data Center at Brookhaven National Laboratory under contract from DOE. The MCNP computer code is used throughout the world and has been extensively tested with its ENDF/B-V-based cross sections. The code development group at LANL, where MCNP was developed, has a set of 25 calculational benchmarks that extensively test various options within the code. Additionally, MCNP Version 4B2 has been certified for use on Waste Management Northwest PC computer platforms (Savino 1998 and McCoy 1998). Section D-6.7 justifies the validity of the calculational methods and neutron cross sections used by reporting the results of critical benchmark experiment analyses.

D-6.4 Single Package Evaluation - Model Description

Section D-6.4.1 describes the single package model for the IF-300 Cask containing SPC PWR fuel assemblies. Section D-6.4.1.1 discusses the results of the single package criticality calculations for the IF-300 Cask containing SPC PWR fuel assemblies.

Section D-6.4.2 addresses the IF-300 Cask loaded with GE BWR fuel assemblies. Section D-6.4.2.1 contains an evaluation of the most reactive BWR fuel design used in CP&L's Brunswick's Power Plant since this analysis addresses five different GE BWR fuel designs (i.e., GE-7, 8, 9, 10, and 13). Section D-6.4.2.2 describes the single package model for the IF-300 Cask loaded with GE BWR fuel assemblies. Section D-6.4.2.3 discusses the results of the single package criticality calculations for the IF-300 Cask containing GE BWR fuel assemblies.

D-6.4.1 IF-300 Cask with SPC PWR Fuel

The Siemens Power Corporation (SPC) 15x15 fuel assembly design incorporates GdO_2 integral burnable absorbers and 6 inch top and bottom axial blankets containing natural UO_2 pellets (0.72 wt% ^{235}U) with a [redacted] theoretical fuel density. The SPC assembly contains 204 fuel rods, 20 guide tubes, and 1 instrument tube. Figures D-6.2-2a and D-6.2-2b illustrate the SPC fuel assembly model used in the evaluation, and Table D-6.2-2 summarizes the SPC fuel design data.

Although the pin-by-pin ^{235}U enrichment varies throughout the fuel lattice in the central enriched zone, all fuel pins in this zone were assumed to contain UO_2 pellets enriched to 4.65 wt% ^{235}U. Consistent with the requirements contained in NUREG/CR- 5661 (NRC 1997), no credit is taken for fuel burnup or integral burnable absorbers such as gadolinium.

The upper (UTP) and lower tie plates (LTP) are modeled as rectangular plates. The thickness of each plate was calculated using the mass and density of the stainless steel (SS) plate and the outer dimensions of the fuel assembly. The outer dimensions of the fuel assembly were used for the tie plates to simplify the model, although this dimension has little impact on the effective multiplication of the system. The masses of the SS tie plates were taken from Table D-6.2-2.

The axial spacers were explicitly modeled as a thin layer of Zr-4 surrounding each of the 15x15 lattice locations. The spacers were modeled as if they extended the full active fuel length (144 inches), although they are actually positioned at 9 distinct axial locations along the length of the fuel assembly with axial spacings from [redacted] in to [redacted] in. The

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thickness of the spacers was calculated using the total weight of 7.78 kg for all of the Zr spacers, the number of lattice elements (15x15=225), the length of the active fuel (144 in), and the Zr density (6.565 g/cc). The total weight of the Zr spacers was taken from Table D-6.2-2. Although this method of modeling is an approximation, it accounts for the total amount of spacer material in the active core region and still maintains the radial heterogeneity of the spacers between the fuel rods.

The PWR basket contains rods containing B_4C for criticality control. These rods are 0.5 in diameter (0.02 in wall) 304 SS tubes filled with B_4C having an average compacted density of 1.76 ± 0.13 g/cc of natural boron with 18.3 ± 0.3 wt% ^{10}B (CSAR 1995). The B_4C rods are positioned between axial spacer disks, and vary in length from 8.75 in to 21.5 in. Although the B_4C rods are actually separated axially by nine 1-inch SS spacer disks, the rods are modeled as if they extended the full length of the basket. The density of the B_4C used in the model for the poison rods is appropriately reduced to account for the spacers and other non-poison material in the poison rods (i.e., end caps, and packing).

The inner cavity height for the PWR basket configuration is 169.5 in and there is a spacer assembly that is placed above the fuel to fill the space between the top of the fuel and the top of the cask cavity. Most of the volume associated with the spacer assembly is void space. Therefore, the MCNP model assumes the spacer assembly is a void area in the containment that is filled with whatever is placed in the containment for the case of interest (e.g., water, mist, air, etc.).

Consistent with the requirements contained in NUREG/CR-5661 (NRC 1997), credit is only taken for 75% of the boron poison present in the IF-300 Cask PWR basket. "A percentage of neutron absorber material greater than 75% may be considered in the analysis only if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented. Limiting added absorber material credit to 75% without comprehensive tests is based on concerns for potential "streaming" of neutrons due to nonuniformities" (NRC 1997). Because comprehensive tests were not performed for the boron poison in the cask basket, credit is only taken for 75% of the boron poison present.

Figure D-6.4-1 illustrates the IF-300 PWR cask model used in the evaluation. The cask was loaded with six SPC fuel assemblies containing UO_2 pellets enriched to 4.65 wt% ^{235}U in the active enriched zone. The center location is left empty for this evaluation. Scoping criticality calculations indicate that the cask will not meet the criticality safety criteria with 6 assemblies at 4.65 wt% ^{235}U if an assembly is loaded in the center PWR basket location and a peripheral basket location is left empty instead of the center location. Note that the cask coolant jacket is not explicitly included in the cask model. The cask coolant jacket encircles the active fuel zone and is a 6 inch water-filled annulus which acts as a neutron shield. Because the model includes a close fitting water reflector around the cask, the neutronic effects of the cask coolant jacket are accounted for in the model.

Table D-6.4-1 lists the material compositions used in the MCNP models of the cask system. Table D-6.4-2 summarizes the PWR cask geometry and material compositions used in the MCNP models. Single cask cases were run for various water densities within the cask containment to determine optimum moderation. Figures D-6.4-1a and D-6.4-1b illustrate the cask model used for these calculations. All cases use a 12 inch close fitting water reflector around the outside of the cask. Sensitivity cases were run at optimum containment moderation to assess the impact of the positioning of the fuel assemblies in the basket on the effective multiplication of the system. The nominal base case is for the fuel assemblies centered in their respective basket locations. A second case was run with the fuel assemblies shifted outward to maximize the assembly-to-assembly spacing. A third case was run with the fuel assemblies shifted inward to minimize the assembly-to-assembly spacing. The dimensions of the IF-300 PWR fuel basket and the fuel assemblies are used as a basis for determining the amount of offset which could be assumed to occur. Manufacturing tolerances for the basket are included in the offset calculation. The total effective fuel assembly offset is calculated to be 0.208 in which is applied simultaneously in the X and Y model directions. Therefore, the fuel assembly positions were shifted by 0.208 in, which represents near contact with the basket. An example MCNP input file is included in Appendix D-6.9.

D-6.4.1.1 Results of the Single Package Evaluation for the IF-300 Cask with SPC PWR Fuel Assemblies from Robinson

10 CFR 71 requires that the packaging be analyzed with the most reactive packaging and content configuration consistent with the effects of normal conditions of transport or hypothetical accident conditions, whichever is more reactive. The existing Consolidated Safety Analysis Report (CSAR 1995) for the IF-300 Cask indicates that the fuel basket (including all structural components, spacer disks, etc.) neutron absorption media, fuel pins and fuel assembly structure survive the NCT and HAC events without significant distortion or permanent deformation. The CSAR also states that the configuration of the fuel and neutron absorption media is maintained to assure criticality safety. Therefore, the NCT and HAC cask models are the same. Section D-6.4.1.1-A addresses the impact of mechanical uncertainties on the criticality safety calculations, and Section D-6.4.1.1-B summarizes the results of the MCNP calculations for the single package evaluation after consideration of all uncertainties.

D-6.4.1.1-A Mechanical Uncertainties

Mechanical uncertainties which may affect the criticality safety of the cask include the effects of variations in moderator density, fuel assembly location, fuel rod pitch, and neutron absorber manufacturing tolerance. Each of these uncertainties is discussed below.

Fuel Assembly Location

The effect of fuel assembly location is addressed by including sensitivity cases with the fuel assemblies in their nominal centered location within the basket, with the assemblies shifted outward, and with the assemblies shifted inward towards the center of the cask. As shown in Table D-6.4-3, the effective multiplication of the cask system was found to be a maximum for the case where all six of the assemblies are shifted radially outward in their respective basket locations.

Fuel Assembly Pin Pitch

Although there is no significant distortion or permanent deformation of the cask/basket/fuel system after HAC, Section A-6.4.3 contains an evaluation of the effect of variations in the fuel assembly pin pitch on the effective multiplication of the system to address deflection of the fuel during accident conditions. The study in CSAR (1995) indicates that there is only a slight increase in the effective multiplication when the pitch is increased, and a relatively large decrease in the effective multiplication when the pitch is decreased. Therefore, it was concluded that no reactivity bias was necessary to account for potential changes in the fuel pin pitch since any accident results in a decrease in the pitch, which decreases the effective multiplication of the system.

Moderator Density

Single cask cases were run for various water densities within the cask containment to determine optimum moderation conditions. The effective multiplication of the cask system was found to be at a maximum for full density water (i.e., 1 g/cc) conditions in the cask containment as shown in Table D-6.4-3. These results are consistent with those found in Section A-6.4.3 which addressed the effects of variations in moderator temperature/density on the reactivity of the system. The results of the moderator temperature/density study in Section A-6.4-3 indicated that the system is most reactive at the maximum water density condition, and that the reactivity falls off monotonically with respect to moderator density. This leads to the observation that the system is undermoderated with respect to overall moderator density. Therefore, no additional bias is necessary to account for moderator temperature/density effects.

Neutron Absorber Manufacturing

The IF-300 PWR fuel basket uses rods containing B_4C for criticality control. The boron density and boron content used in the models are considered worst case, i.e., minimum compacted density (1.63 g/cc) and ^{10}B enrichment (18 wt%). Additionally, credit is only taken for 75% of the boron poison present per NUREG (1997), therefore, no additional neutron absorber manufacturing uncertainties or biases need to be accounted for in the models.

D-6.4.1.1-B Summary of Single Package Evaluation Results for the IF-300 Cask with SPC PWR Fuel

Table D-6.4-3 summarizes the results of the calculations made for the single IF-300 cask with a 12 in close-fitting water reflector around the cask. Most of the MCNP cases were run using 3000 neutrons per cycle for a total of 220 cycles with the first 30 cycles being skipped to ensure that the source distribution was reasonably converged. The maximum k_{eff} including bias and uncertainties is 0.94689 and occurs for full density water conditions in the cask containment with the fuel assemblies shifted outward to maximize the assembly-to-assembly spacing. Note that the effective multiplication of the system decreases rapidly with decreasing containment water density.

An additional MCNP case was run with 6 SPC PWR fuel assemblies, however, in this case an assembly was loaded in the center PWR basket location and a peripheral basket location was left empty instead of the center location. This calculation indicates that the cask will not meet the criticality safety criteria if an assembly is placed in the center location and 6 assemblies at 4.65 wt% ^{235}U are loaded into the cask.

It is concluded that a single IF-300 Cask will meet the criticality safety limit of $k_{eff} < 0.95$ when the cask contains six SPC PWR fuel assemblies with a maximum lattice average enrichment of 4.65 wt% ^{235}U loaded in the peripheral PWR basket locations (i.e., with the center basket location empty). Therefore, the IF-300 Cask meets the requirement that a single package must be subcritical if water were to leak into the containment system under the following conditions:

- Most reactive credible configuration consistent with the chemical and physical form of the material.

- Moderation by water to the most reactive credible extent.

- Close full reflection of the containment system by water on all sides or such greater reflection of the system as may be provided by the material of the packaging.

This conclusion is reached with the following conservative assumptions:

1) The fuel is unirradiated (i.e., has 0 MWD/MTU burnup).
2) No credit is taken for the gadolinia present in the fuel before irradiation.
3) Credit is only taken for 75% of the poison present in the basket.

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure D-6.4-1a

PWR Single Cask Model - Radial View

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure D-6.4-1b

PWR Single Cask Model - Axial View

Table D-6.4-1

Material Compositions Used in the MCNP Model for the IF-300 PWR Cask (2 sheets total)

Material	g/cc	Elem	wt%
UO2 fuel - 4.65 wt% U235	10.4	U-235	4.0987
		U-238	84.0458
		O	11.8554
He (gap between pellet and clad)[a]	1.78E-04	He	1.0000
Zircaloy 4	6.565	Zr	98.1800
		O	0.1200
		Cr	0.1000
		Fe	0.2000
		Sn	1.4000
Water	1[b]	H	11.1111
		O	88.8889
Natural UO2 blanket - 0.72 wt% U235	10.4	U-235	0.6267
		U-238	87.5206
		O	11.8528
304L SS	8.03	C	0.0300
		Si	1.0000
		P	0.0450
		S	0.0300
		Cr	20.0000

Material	g/cc	Elem	wt%
		Mn	2.0000
		Fe	64.8950
		Ni	12.0000
B4C poison	1.63[c]	B-10	14.0985
		B-11	64.2267
		C	21.6748
Depleted Uranium	18.82	U-235	0.2200
		U-238	99.7800
Channel - 304 SS/water (VF water = 0.358)	5.513[d]	H	0.7266
		C	0.0748
		O	5.7669
		Si	0.9351
		P	0.0421
		S	0.0281
		Cr	18.7013
		Mn	1.8701
		Fe	62.0369
		Ni	9.8182

[a] The gap between the fuel pellet and clad normally contains helium, however, the model conservatively assumed the gap was flooded with water which was found to be a more reactive condition.

[b] The density of the water in the cask varies for different cases.

[c] The poison pin B_4C density listed on the PWR basket drawing (Drawing 159C5238 - Sheet 6, Rev 8 - page 4-16) is specified as 1.76 +/- 0.13 g/cc. The value listed is for the lower end of the tolerance band. The actual B_4C density used in the MCNP model is significantly less since credit may only be taken for 75% of the poison, and the density is further reduced to account for the pins being modeled the full length of the basket.

[d] The SS/water mixture for the channel assumes the cask is flooded with water at a density of 1 g/cc. The density and wt % for the SS/water mixture will vary as a function of the density of the water in the cask.

Table D-6.4-2

PWR Cask Geometry and Compositions used in the MCNP Models

Axial Regions		
	Zone Thickness	
Zone Material	In	Cm
Top of Cask		
304 SST	1.5	3.81
Cast DU	3.75	9.525
304 SS	1.25	3.175
Cask Inner Cavity Height[a]	180.25	457.84
304 SST	1	2.54
Cast DU	3	7.62
304 SST	2	5.08
Bottom of Cask		
Radial Regions		
Cask Inner Cavity Radius	18.75	47.625
317 SS Inner Shell	0.5	1.27
Cast DU	4	10.16
317 SS Outer Shell	1.56	3.9624

[a] The inner cavity height for the PWR basket configuration is 169.5 in, and there is a spacer assembly that is placed above the fuel to fill the space between the top of the fuel and the top of the cask cavity. The MCNP model assumes the spacer assembly is a void area in the containment that is filled with whatever is placed in the containment for the case of interest (e.g., water, mist, air, etc.). Therefore, the cask cavity is modeled with a height of 180.25 in.

Table D-6.4-3

Effective Multiplication Factors for a Single IF-300 PWR Cask[a] With 6 SPC Fuel Assemblies at 4.65 wt% ^{235}U

MCNP Case	Containment Water Density	k_{eff}	σ	k_{tot}[b]
Ifp6-10	1	0.92738	0.00114	0.94215
Psi6-10[c]	1	0.92496	0.00103	0.93966
Pso6-10[c]	1	0.93221	0.00099	0.94689
Ifp6-98	0.98	0.92148	0.00103	0.93618
Ifp6-96	0.96	0.91259	0.00113	0.92735
Ifp6-94	0.94	0.90337	0.00117	0.91816
Ifp6-90	0.9	0.8901	0.00154	0.90515
Ifp6-70	0.7	0.81105	0.00164	0.82618
Ifp6-40	0.4	0.67279	0.00278	0.68913
Ifp6-01	0.1	0.47494	0.00165	0.49008
Ifp6-00	0.0	0.33435	0.00121	0.34916

[a] Credit is only taken for 75% of the ^{10}B poison present in the cask basket.
[b] k_{tot} = k_{eff} including bias and uncertainties (see Section D-6.7).
[c] Run pso6-10 shifts the fuel assemblies outward in the basket to maximize the assembly-to-assembly spacing. Run psi6-10 shifts the fuel assemblies inward in the basket to minimize the assembly-to-assembly spacing. The remaining cases have the assemblies in their nominal (centered) positions.

D-6.4.2 IF-300 Cask with GE BWR Fuel

Section D-6.4.2.1 contains an evaluation of the most reactive BWR fuel design used in CP&L's Brunswick's Power Plant since this analysis addresses five different GE BWR fuel designs (i.e., GE-7, 8, 9, 10, and 13). Section D-6.4.2.2 describes the single package model for the IF-300 Cask loaded with GE BWR fuel assemblies. Section D-6.4.2.3 discusses the results of the single package criticality calculations for the IF-300 Cask containing GE BWR fuel assemblies.

D-6.4.2.1 <u>Evaluation of Most Reactive BWR Fuel Design used in CP&L's Brunswick Power Plant</u>

There are 5 different General Electric (GE) BWR fuel designs from CP&L's Brunswick Nuclear Power Plant that are intended to be shipped in the IF-300 BWR Cask. These fuel designs are identified as GE-7, GE-8, GE-9, GE-10, and GE-13. All of these fuel designs have 8x8 lattices of fuel rods except for GE-13, which has a 9x9 lattice. Table D-6.2-3 provides further details on the dimensions and compositions for the different fuel designs. These five GE BWR fuel designs were evaluated to determine which design was the most reactive.

MCNP calculations were performed for each of the five GE fuel designs using mirror reflective boundary conditions in the radial direction for a fuel assembly with its fuel channel and a water gap outside the channel. A 12 inch water reflector on the top and bottom of the fuel assembly was used in the model. This model represents an infinite square array of fuel assemblies. The thickness of the water gap outside the channel in the infinite array was selected to simulate the spacing of the fuel assemblies in the cask, which is approximately 7 inches. Because the cask contains an array of 17 fuel assemblies, the results from these runs will provide an indication of which fuel assembly design is the most reactive.

The GE-7 and 8 fuel was modeled with 6 inch top and bottom axial blankets containing natural UO_2 pellets (0.72 wt% ^{235}U). The GE-9 and 10 fuel was modeled with 12 inch top and 6 inch bottom axial blankets containing natural UO_2 pellets. The GE-13 fuel was modeled with an 8 inch top and 6 inch bottom axial blanket. Although the pin-by-pin ^{235}U enrichment varies throughout the fuel lattice in the central enriched zone, all fuel pins in this zone were assumed to contain UO_2 pellets enriched to 4.25 wt% ^{235}U. CP&L indicated that this is the maximum lattice average enrichment for the fuel designs to be shipped in the IF-300 BWR cask. Note that the GE-13 fuel design contains partial length fuel rods in various fuel lattice locations. These fuel rods extend from the lower tie plate (LTP) up to the 6th axial spacer grid. For simplicity, this analysis assumes the partial length fuel rods are identical to the normal fuel rods. This results in a fuel assembly that contains slightly more enriched uranium than exists in reality, which is conservative. Consistent with the requirements contained in NUREG/CR-5661 (NRC 1997), no credit is taken for fuel burnup or integral burnable absorbers such as gadolinium.

The upper (UTP) and lower tie plates (LTP) are modeled as rectangular plates. The thickness of each plate was calculated using the mass and density of the stainless steel (SS) plate and the outer dimensions of the fuel assembly. The outer dimensions of the fuel assembly was used for the tie plates to simplify the model, although this dimension has little impact on the effective multiplication of the system. The masses of the SS tie plates were taken from Table D-6.2-3.

The axial spacers were explicitly modeled as a thin layer of Zr-2 or Zr-4 (see Table D-6.2-3) surrounding each of the 8x8 or 9x9 lattice locations. The spacers were modeled as if they extended the full active fuel length (150 or [redacted] inches). The thickness of the spacers was calculated using the total weight for all of the Zr spacers (Table D-6.2-3), the number of lattice elements (appropriately reduced for large water holes in GE 9, 10, & 13), the length of the active fuel, and the Zr density (6.565 g/cc). Note that the weight of inconel in the active region was not included in the calculation of the spacer thickness. The total weight of the Zr spacers for each fuel type was taken from Table D-6.2-3. Although this method of modeling is an approximation, it accounts for the total amount of spacer material in the active core region and still maintains the radial heterogeneity of the spacers between the fuel rods. Table D-6.4-5 lists the material compositions used in

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The highlighted data is proprietary information to General Electric.

the MCNP models of the infinite array of fuel assemblies. Table D-6.4-6 summarizes the cask geometry and material composition used in the MCNP models.

The infinite array cases for the five GE fuel designs were run in MCNP using 1000 neutrons per cycle for a total of 200 cycles. The first 20 cycles were skipped to ensure that the source distribution was reasonably converged. The MCNP runs produced what are essentially k_∞ values for each fuel design since the model used mirror reflective boundary conditions in the radial direction and a 12 inch reflector in the axial directions. Calculations were performed for a water density of 1 g/cc. This water density was selected because preliminary calculations for the cask indicated that the effective multiplication of the cask system is maximized when the cask containment is flooded with full density water (1 g/cc). Calculations were made for both ends of the fuel density range (94.4% - 98% TD) to assess the impact of the fuel density on the effective multiplication of the infinite array of fuel assemblies. An example MCNP input file for the GE-8 fuel type is included in Section D-6.9-Appendix.

The results in Table D-6.4-4 indicate that the GE-8 fuel has the highest k_∞ value, and this occurs for the higher end of the fuel density range (98% TD). The differences between the higher k_∞ values are statistically insignificant, which indicates that the reactivity of these fuel designs are nearly the same. However, the GE-8 fuel with a 98% TD has the highest k_∞ value, therefore, this fuel design will be used for the BWR cask criticality calculations.

Table D-6.4-4

Infinite Multiplication Factors for GE Fuel Designs

Fuel Type	%TD	k_∞	σ	MCNP input file
GE-7	94.4	1.19818	0.00133	GE7INF
GE-7	98	1.20004	0.00132	GE7INA
GE-8	95	1.19777	0.00138	GE8INF
GE-8	98	**1.20105**	0.00139	GE8INA
GE-9	94.4	1.19551	0.00138	GE9INF
GE-9	98	1.19995	0.00147	GE9INA
GE-10	94.4	1.19703	0.00140	GE10IN
GE-10	98	1.19772	0.00131	GE10IA
GE-13	94.4	1.19555	0.00137	GE13IN
GE-13	98	1.19761	0.00132	GE13IA

D-6.4.2.2 IF-300 Cask With GE BWR Fuel

Figures D-6.4-2 and D-6.4-3 illustrate the IF-300 BWR cask model used in the evaluation. The cask was loaded with 17 channeled GE-8 fuel assemblies containing UO_2 pellets enriched to 4.25 wt% ^{235}U in the active enriched zone. The GE-8 fuel assembly design was found to be the most reactive, as discussed in Section D-6.4.2.1.

The BWR basket contains borated stainless steel plates for criticality control. The material properties for these plates are as follows (CSAR 1995):

Finished plate thickness	0.250" to 0.375" +0.045"/-0.010" Long/Medium Horizontal- 0.25" (see Figure D-6.4-2) Short Vertical- 0.375" (see Figure D-6.4-2)

Widths of plates	Long Length Horizontal - 33" Medium Length Horizontal - 26" Short Vertical next to center assembly - 6.63" Short Vertical below center assembly - 6.375"
SS304 "Skin Thickness"	0.007" maximum, each side
Boron Content	1.0 w/o minimum
B-10 Enrichment	94% (by atoms) B-10, min
Plate density	7.76 g/cc minimum

Consistent with the requirements contained in NUREG/CR-5661 (NRC 1997), credit is only taken for 75% of the boron poison present in the IF-300 Cask BWR basket. "A percentage of neutron absorber material greater than 75% may be considered in the analysis only if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented. Limiting added absorber material credit to 75% without comprehensive tests is based on concerns for potential "streaming" of neutrons due to nonuniformities" (NRC 1997). Because comprehensive tests were not performed for the boron poison in the cask basket, credit is only taken for 75% of the boron poison present.

The BWR basket for the 17 channeled fuel assemblies consists of nine 2 inch SS axial spacers with 17 square holes for the channeled fuel assemblies to fit into. The borated SS poison plates are positioned vertically between the 2 inch axial spacers. Therefore, the poison plates do not extend the full length of the basket. The 2 inches of steel for the spacers is explicitly modeled between the poison plates in the axial direction; however, the rest of the steel for the spacers in the radial direction is not included in the model. This steel is effectively replaced by water or whatever is assumed to fill the cask (e.g., air for a dry cask). Table D-6.4-5 lists the material compositions used in the MCNP models of the cask system. Table D-6.4-6 summarizes the cask geometry and compositions used in the MCNP models. Note that the cask coolant jacket is not explicitly included in the cask model. The cask coolant jacket encircles the active fuel zone and is a 6 inch water-filled annulus which acts as a neutron shield. Because the model includes a close fitting water reflector around the cask, the neutronic effects of the cask coolant jacket are accounted for in the model.

Single cask cases were run for various water densities within the cask containment to determine optimum moderation. Figures D-6.4-2 and D-6.4-3 illustrate the cask model used for these calculations. All cases use a 12 inch close fitting water reflector around the outside of the cask. Sensitivity cases were run at optimum containment moderation to assess the impact of the positioning of the fuel assemblies in the basket on the effective multiplication of the system. The nominal base case is for the fuel assemblies centered in their respective basket locations. A second case was run with the fuel assemblies shifted outward to maximize the assembly-to-assembly spacing. A third case was run with the fuel assemblies shifted inward to minimize the assembly-to-assembly spacing. Dimensions of the IF-300 Channeled BWR fuel basket and the fuel assemblies are used as a basis for determining the amount of offset which could be assumed to occur. A stack of manufacturing tolerances is included in the offset calculation. The total effective fuel assembly offset is calculated to be 0.268 in which is applied simultaneously in the X and Y model directions. Therefore, the fuel assembly positions were shifted by 0.268 in, which represents near contact with the basket.

D-6.4.2.3 <u>Results of the Single Package Evaluation for the IF-300 Cask with GE BWR Fuel Assemblies from Brunswick</u>

10 CFR 71 requires that the packaging be analyzed with the most reactive packaging and content configuration consistent with the effects of normal conditions of transport or hypothetical accident conditions, whichever is more reactive. The existing Consolidated Safety Analysis Report (CSAR 1995) for the IF-300 Cask indicates that the fuel basket (including all structural components, spacer disks, etc.) neutron absorption media, fuel pins and fuel assembly structure survive the NCT and HAC events without

significant distortion or permanent deformation. The CSAR also states that the configuration of the fuel and neutron absorption media is maintained to assure criticality safety. Therefore, the NCT and HAC cask models are the same. Section D-6.4.2.3-A addresses the impact of mechanical uncertainties on the criticality safety calculations, and Section D-6.4.2.3-B summarizes the results of the MCNP calculations for the single package evaluation after consideration of all uncertainties.

D-6.4.2.3-A <u>Mechanical Uncertainties</u>

Mechanical uncertainties which may affect the criticality safety of the cask include the effects of variations in moderator density, fuel assembly location, fuel rod pitch, and neutron absorber manufacturing tolerance. Each of these uncertainties is discussed below.

<u>Fuel Assembly Location</u>

The effect of fuel assembly location is addressed by including sensitivity cases with the fuel assemblies in their nominal centered location within the basket, with the assemblies shifted outward, and with the assemblies shifted inward towards the center of the cask. The effective multiplication of the cask system was found to be a maximum for the case where all of the assemblies are shifted radially inward in their respective basket locations, as shown in Table D-6.4-7.

<u>Fuel Assembly Pin Pitch</u>

Although there is no significant distortion or permanent deformation of the cask/basket/fuel system after HAC, Section A-6.4.3 contains an evaluation of the effect of variations in the fuel assembly pin pitch on the effective multiplication of the system to address deflection of the fuel during accident conditions. The study in CSAR (1995) indicates that there is only a slight increase in the effective multiplication when the pitch is increased, and a relatively large decrease in the effective multiplication when the pitch is decreased.

Therefore, it was concluded that no reactivity bias was necessary to account for potential changes in the fuel pin pitch since any accident results in a decrease in the pitch, which decreases the effective multiplication of the system.

Moderator Density

Single cask cases were run for various water densities within the cask containment to determine optimum moderation conditions. The effective multiplication of the cask system was found to be at a maximum for full density water (i.e., 1 g/cc) conditions in the cask containment, as shown in Table D-6.4-7. These results are consistent with those found in Section A-6.4.3 which addressed the effects of variations in moderator temperature/density on the reactivity of the system. The results of the moderator temperature/density study in Section A-6.4-3 indicate that the system is most reactive at the maximum water density condition, and that the reactivity falls off monotonically with respect to moderator density. This leads to the observation that the system is undermoderated with respect to overall moderator density. Therefore, no additional bias is necessary to account for moderator temperature/density effects.

Neutron Absorber Manufacturing.

The IF-300 Channeled BWR fuel basket design uses NeutroSorb Plus™ as a fixed neutron absorber which is required to achieve criticality control for the design basis fuel.

Sensitivity calculations are performed to determine the impact of absorber plate thickness variations on the effective multiplication of the system. All other parameters are considered to be worst case (maximum skin thickness, minimum boron content at minimum B10 enrichment, minimum plate density). No parametric studies were run for boron content or isotopic enrichment since those parameters are specified as minimums and since they do not affect the overall fuel/moderator volume ratio. Additionally, credit is only taken for 75% of the boron poison present per NUREG (1997).

It is determined that a reduction of 0.01 inch the plate thickness to account for manufacturing tolerance presented the worst case for criticality. The effective multiplication of the system with the nominal thickness neutron absorber is calculated to be:

$$k_{eff} = 0.90239 \pm 0.00082$$

The effective multiplication of the system with the minimum thickness poison plates is calculated to be:

$$k_{eff} = 0.90471 \pm 0.00086$$

This results in a bias of 0.00232 with a one sigma uncertainty of 0.00086 to address the manufacturing tolerance associated with the thickness of the poison plates.

D-6.4.2.3-B <u>Summary of Single Package Evaluation Results for the IF-300 Cask with GE BWR Fuel</u>

Table D-6.4-7 summarizes the results of the calculations made for the single IF-300 cask with a 12 in close-fitting water reflector around the cask. Most of the MCNP cases were run using 3000 neutrons per cycle for a total of 200 cycles with the first 30 cycles being skipped to ensure that the source distribution was reasonably converged. The maximum k_{eff} of 0.91473 (σ =0.00083) occurs for full density water conditions in the cask containment with the fuel assemblies shifted inward to minimize the assembly-to-assembly spacing. Note that the effective multiplication of the system decreases rapidly with decreasing containment water density. The k_{tot} including the bias and uncertainties from the benchmark experiments (see Section D-6.7), and the bias and uncertainty associated with the tolerance for the thickness of the boron plates are β_{plate} = 0.93185 and σ_{plate} = 0.00086, respectively, as shown below.

$$k_{tot} = k_{eff} + \beta_{benchmark} + \beta_{plate} + 2(\sigma^2 + \sigma^2_{crit} + \sigma^2_{exp} + \sigma^2_{plate})^{0.5}$$

or

$$k_{tot} = 0.91473 + 0.0074 + 0.00232 + 2(0.00083^2 + 0.00316^2 + 0.00151^2 + 0.00086^2)^{0.5}$$

$$k_{tot} = 0.93185$$

It is concluded that a single IF-300 Cask will meet the criticality safety limit of $k_{eff} < 0.95$ when the cask contains seventeen channeled BWR fuel assemblies with a maximum lattice average enrichment of 4.25 wt% ^{235}U. Therefore, the IF-300 Cask meets the requirement that a single package must be subcritical if water were to leak into the containment system under the following conditions:

- Most reactive credible configuration consistent with the chemical and physical form of the material.

- Moderation by water to the most reactive credible extent.

- Close full reflection of the containment system by water on all sides or such greater reflection of the system as may be provided by the material of the packaging.

This conclusion is reached with the following conservative assumptions:

1) The fuel is unirradiated (i.e., has 0 MWD/MTU burnup).
2) No credit is taken for the gadolinia present in the fuel before irradiation.
3) Credit is only taken for 75% of the poison present in the basket.

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure D-6.4-2

BWR Cask Model - Radial View

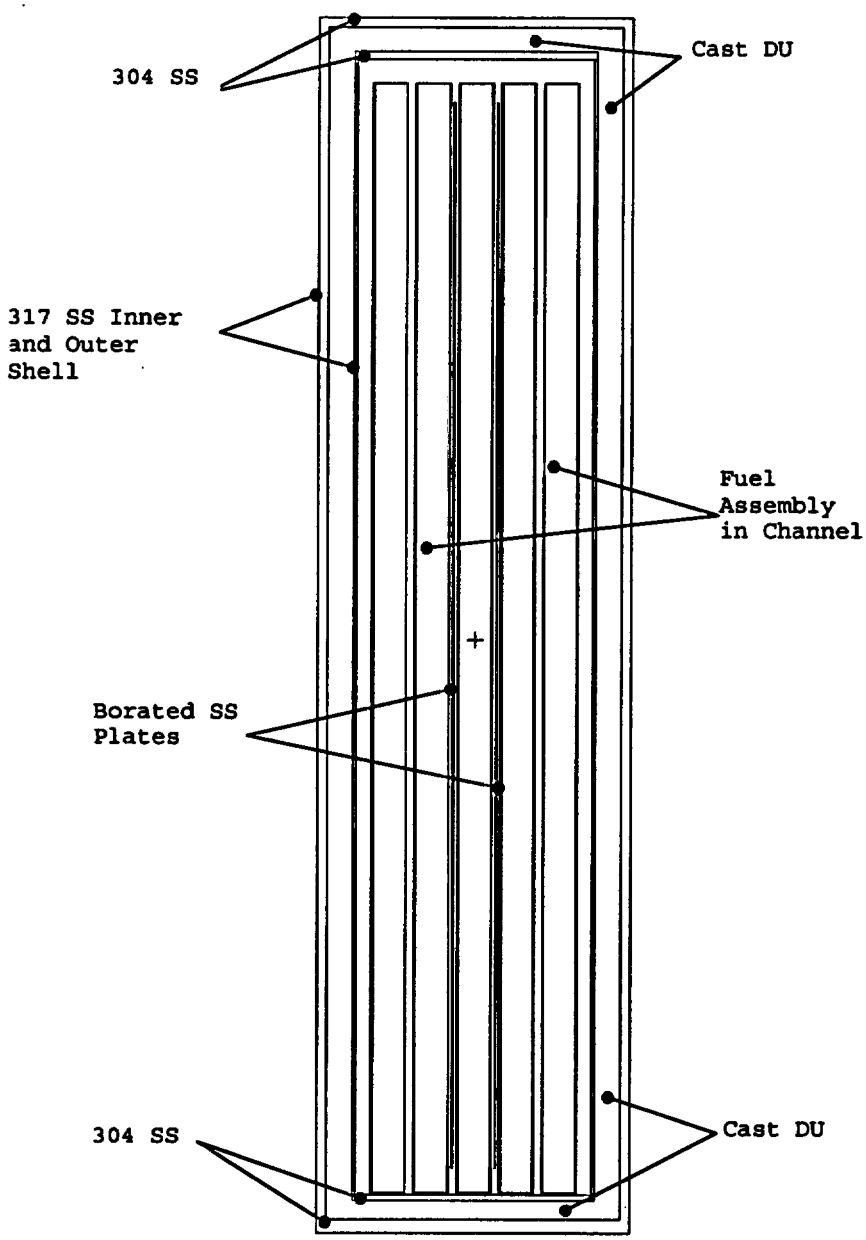

Figure D-6.4-3

BWR Cask Model - Axial View

Table D-6.4-5

Material Compositions Used in the MCNP Model for the GE-8 Fuel And the IF-300 BWR Cask (2 sheets total)

Material	g/cc	Elem	wt%
UO2 fuel - Enriched to 4.25 wt% U235 (94.4%-98% TD)	10.346-10.74	U-235	3.7461
		U-238	84.3965
		O	11.8575
Natural UO2 blanket - 0.72 wt% U235	10.346-10.74	U-235	0.6267
		U-238	87.5206
		O	11.8528
He (gap between pellet and clad)[a]	1.78E-04	He	1.0000
Zircaloy 2	6.565	Zr	98.2300
		O	0.1200
		Cr	0.1000
		Fe	0.1000
		Ni	0.0500
		Sn	1.4000
Zircaloy 4	6.565	Zr	98.1800
		O	0.1200
		Cr	0.1000
		Fe	0.2000
		Sn	1.4000

Material	g/cc	Elem	wt%
water	1[b]	H	11.1111
		O	88.8889
304L SS	8.03	C	0.0300
		Si	1.0000
		P	0.0450
		S	0.0300
		Cr	20.0000
		Mn	2.0000
		Fe	64.8950
		Ni	12.0000
Depleted Uranium	18.82	U-235	0.2200
		U-238	99.7800
Neutrosorb Poison Plates	7.739[c]	B-10	0.7069[c]
		B-11	0.0606
		Cr	19.0452
		Mn	1.0024
		Fe	65.6558
		Ni	13.5295

[a] The gap between the fuel pellet and clad normally contains helium, however, the model conservatively assumed the gap was flooded with water.

[b] The density of the water in the cask varies for different cases.

[c] The density of the Neutrosorb poison plates is specified as a minimum of 7.76 g/cc. The value listed for the density and wt% ^{10}B accounts for the fact that credit may only be taken for 75% of the ^{10}B poison.

Table D-6.4-6

BWR Cask Geometry and Compositions used in the MCNP Models

Axial Regions		
	Zone Thickness	
Zone Material	In	cm
Top of Cask		
304 SST	1.5	3.81
Cast DU	3.75	9.525
304 SS	1.25	3.175
Cask Inner Cavity Height	180.25	457.835
304 SST	1	2.54
Cast DU	3	7.62
304 SST	2	5.08
Bottom of Cask		
Radial Regions		
Cask Inner Cavity Radius	18.75	47.625
317 SS Inner Shell	0.5	1.27
Cast DU	4	10.16
317 SS Outer Shell	1.56	3.9624

Table D-6.4-7

Effective Multiplication Factors for a Single IF-300 BWR Cask[a] with 17 GE-8 Fuel Assemblies at 4.25 wt% ^{235}U

MCNP Case	Water Density	k_{eff}	σ	k_{tot}[b]
Ifb810	1	0.90239	0.00082	0.91951
Bsi810[c]	1	0.91473	0.00083	0.93185
Bso810[c]	1	0.88675	0.00089	0.90390
Ifb898	0.98	0.90067	0.00079	0.91777
Ifb896	0.96	0.89619	0.00134	0.91360
Ifb894	0.94	0.89435	0.00184	0.91217
Ifb892	0.92	0.88550	0.00143	0.90298
Ifb870	0.7	0.83637	0.00201	0.85435
Ifb840	0.4	0.70524	0.00180	0.72302
Ifb8010	0.1	0.43769	0.00147	0.45520
Ifb800	0.0	0.32264	0.00099	0.33984

[a] Credit is only taken for 75% of the ^{10}B poison present in the cask basket.

[b] k_{tot} = k_{eff} including the benchmark experiment bias and uncertainties (see Section D-6.7), and the bias and uncertainty associated with the tolerance for the thickness of the boron plates (see Section D-6.4.2.3-A).

[c] Run bso810 shifts the fuel assemblies outward in the basket to maximize the assembly-to-assembly spacing. Run bsi810 shifts the fuel assemblies inward in the basket to minimize the assembly-to-assembly spacing.

D-6.5 Evaluation of Package Arrays Under Normal Conditions of Transfer

Section D-6.5.1 describes the NCT package array model for the IF-300 Cask containing SPC PWR fuel assemblies. Section D-6.5.1.1 discusses the results of the NCT infinite array criticality calculations for the IF-300 Cask containing SPC PWR fuel assemblies.

Section D-6.5.2 describes the NCT package array model for the IF-300 Cask containing GE BWR fuel assemblies. Section D-6.5.2.1 discusses the results of the NCT infinite array criticality calculations for the IF-300 Cask containing GE BWR fuel assemblies.

D-6.5.1 Arrays of IF-300 Casks with PWR Fuel under NCT

An infinite array of casks was modeled to conservatively represent 5N undamaged packages under NCT. A triangular array configuration was used since it results in a tightly-packed array, and the cask pitch was varied to assess the impact of the distance between the casks on the effective multiplication of the cask system. Mirror reflective boundary conditions were used for eight surfaces surrounding the cask. The eight surfaces consist of six side surfaces in the shape of a hexagon, plus the top and bottom surfaces. This model represents an infinite array of casks in both the radial and axial directions. Figure D-6.5-1 illustrates the hexagonal surfaces used for the infinite array models. The cask containment was assumed to be dry, because the casks are shipped dry and there is no leakage into the package, including the containment system, when subject to the NCT tests of 10 CFR 71.71. The area between the casks was modeled as air because an infinite array of casks was modeled and 10 CFR 71.59 states that there is nothing between the packages. As mentioned in Section D-6.4.1, the cask coolant jacket, which is not included in the model, normally contains water which acts as a neutron shield. Although this would provide moderating material between the casks during NCT, the effective multiplication for the cask when the cask containment is dry is so low (see Table D-6.4-8) that the effective neutron multiplication will remain far below the limit of 0.95 when the array of dry casks has water interspersed between them.

D-6.5.1.1 <u>Results of the Infinite Array Package Evaluation for the IF-300 Cask with SPC PWR Fuel Assemblies from Robinson</u>

Table D-6.5-1 summarizes the results of the calculations made for the infinite array of IF-300 casks containing SPC PWR fuel assemblies which conservatively represents 5N undamaged packages under NCT. Due to the extremely low effective multiplication of the NCT infinite cask array system, only 3 different cask pitches were investigated. These cases were run with the assemblies in their nominal (centered) positions. These calculations demonstrate that the casks are effectively isolated from each other. The highest k_{tot} plus bias and uncertainty is 0.38870 and this occurs for an array of casks with a distance of 2 cm between the casks, although all of the values are within the statistical uncertainties of the calculation. An example MCNP input file is included in the Section D-6.9.

It is concluded that an infinite array of IF-300 Casks will meet the criticality safety limit of $k_{eff} < 0.95$ when the cask contains six SPC PWR fuel assemblies with a maximum lattice average enrichment of 4.65 wt% ^{235}U loaded in the peripheral PWR basket locations (i.e., with the center basket location empty). Therefore, the IF-300 Cask meets the requirement that an array of 5 times N undamaged packages stacked together in any arrangement with nothing between the packages must be subcritical, where in this case N is equal to infinity. Note that the model did not use close full reflection of the array on all sides by water because an infinite array was analyzed.

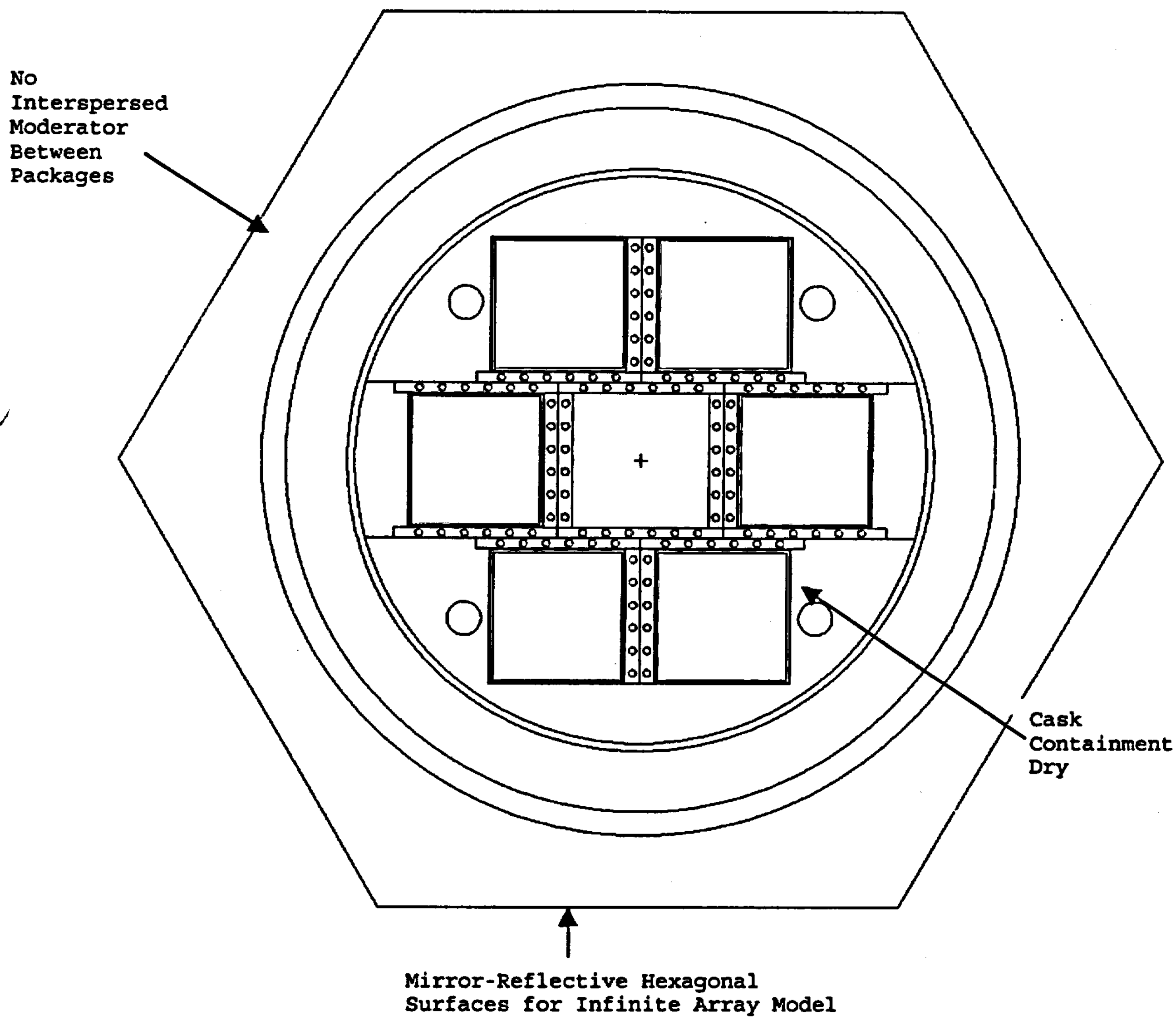

Figure D-6.5-1

PWR Cask NCT Infinite Array Model

Table D-6.5-1

Multiplication Factors for an Infinite Array of IF-300 PWR Casks[a] Under NCT with 6 SPC Fuel Assemblies with 4.65 wt% ^{235}U

MCNP Case	Cask Spacing, cm	k_∞	σ	k_{tot}[b]
ifp6i00a	2	0.37378	0.00137	**0.38870**
ifp6i00c	10	0.37095	0.00151	0.38598
ifp6i00e	20	0.37129	0.00106	0.38601

[a] Credit is only taken for 75% of the ^{10}B poison present in the cask basket.
[b] $k_{tot} = k_\infty$ including bias and uncertainties (see Section D-6.7).

D-6.5.2 Arrays of IF-300 Casks with BWR Fuel under NCT

An infinite array of casks was modeled to conservatively represent 5N undamaged packages. A triangular array configuration was used since it results in a tightly-packed array, and the cask pitch was varied to assess the impact of the distance between the casks on the effective multiplication of the cask system. Mirror reflective boundary conditions were used for eight surfaces surrounding the cask. The eight surfaces consist of six side surfaces in the shape of a hexagon, plus the top and bottom surfaces. This model represents an infinite array of casks in both the radial and axial directions. Figure D-6.5-2 illustrates the hexagonal surfaces used for the infinite array models. The cask containment was assumed to be dry, because the casks are shipped dry and there is no leakage into the package, including the containment system, when subject to the NCT tests of 10 CRF 71.71. The area between the casks was modeled as air because an infinite array of casks was modeled and 10 CFR 71.59 states that there is nothing between the packages. As mentioned in Section D-6.4.1, the cask coolant jacket, which is not included in the model, normally contains water which acts as a neutron shield. Although this would provide moderating material between the casks during NCT, the effective multiplication for the cask when the cask containment is dry is so low (see Table D-6.4-8) that the effective neutron multiplication will remain far below the limit of 0.95 when the array of dry casks has water interspersed between them.

D-6.5.2.1 <u>Results of the Infinite Array Package Evaluation for the IF-300 Cask with GE BWR Fuel Assemblies from Brunswick</u>

Table D-6.5-2 summarizes the results of the calculations made for an infinite array of BWR casks, which conservatively represents 5N undamaged packages. The cask pitch was varied to evaluate the effect of the distance between the casks on the effective multiplication of the cask system. These cases were run with the assemblies in their nominal (centered) position in the basket. The highest k_∞ is 0.35512 and occurs for an array of casks with a distance of 6 cm between the casks, although all of the values are within the statistical uncertainties of the calculation. Because of the extremely low effective multiplication for the infinite array of dry casks, only 3 different cask spacings were investigated. The k_{tot} including the bias and uncertainties from the benchmark experiments (see Section D-6.7), and the bias and uncertainty associated with the tolerance for the thickness of the boron plates is calculated to be 0.37240.

It is concluded that an infinite array of IF-300 Casks will meet the criticality safety limit of $k_{eff} < 0.95$ when the cask contains seventeen channeled BWR fuel assemblies with a maximum lattice average enrichment of 4.25 wt% ^{235}U. Therefore, the IF-300 Cask meets the requirement that an array of 5 times N undamaged packages stacked together in any arrangement with nothing between the packages must be subcritical, where in this case N is equal to infinity. Note that the model did not use close full reflection of the array on all sides by water because an infinite array was analyzed.

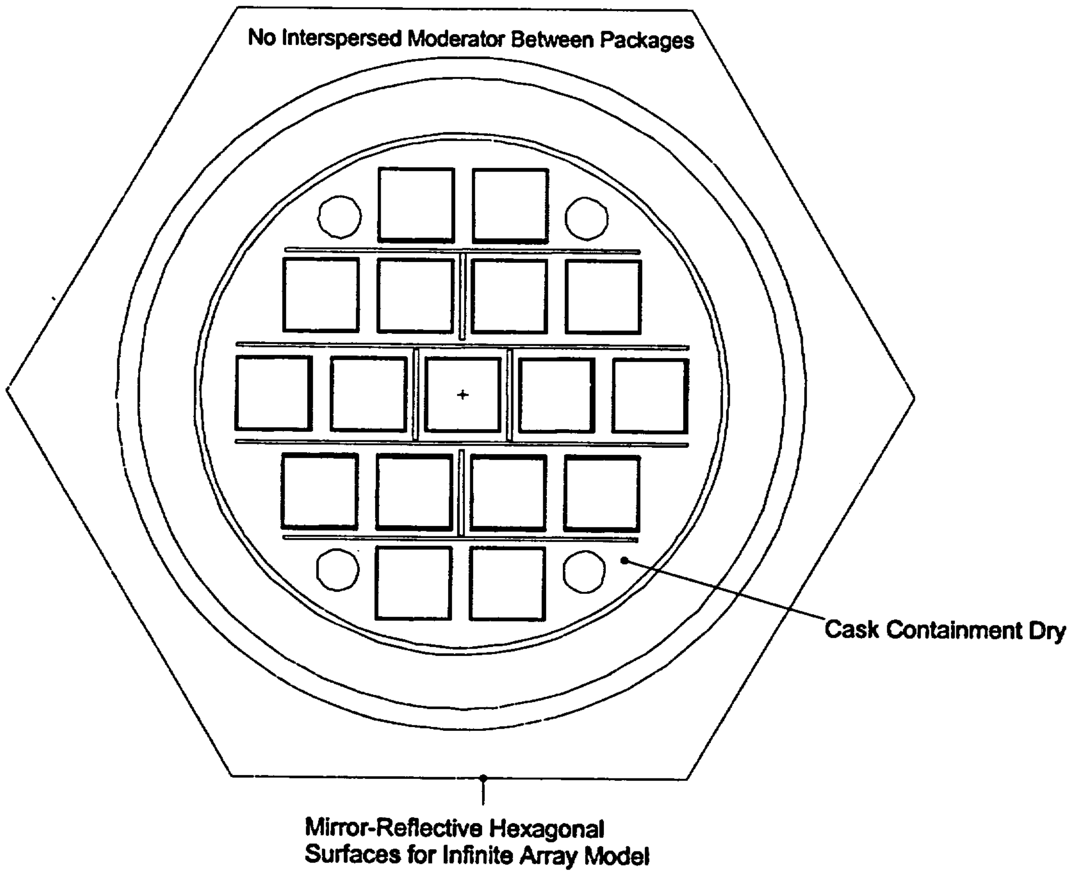

Figure D-6.5-2

BWR Cask NCT Infinite Array Model

Table D-6.5-2

Infinite Multiplication Factors for an Infinite Array of IF-300 BWR Casks[a] Under NCT with 17 GE-8 Fuel Assemblies with 4.25 wt% ^{235}U

MCNP input file	Cask spacing Cm	K_∞	σ	k_{tot}[b]
Ifb8HX00	2	0.35414	0.00114	0.37142
Ifb8HX0a	6	0.35512	0.00114	0.37240
Ifb8HX0b	10	0.35445	0.00114	0.37173

[a] Credit is only taken for 75% of the ^{10}B poison present in the cask basket.

[b] k_{tot} = k_{eff} including the benchmark experiment bias and uncertainties (see Section D-6.7), and the bias and uncertainty associated with the tolerance for the thickness of the boron plates (see Section D-6.4.2.3-A).

D-6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

Section D-6.6.1 describes the HAC package array model for the IF-300 Cask containing SPC PWR fuel assemblies. Section D-6.6.1.1 discusses the results of the HAC infinite array criticality calculations for the IF-300 Cask containing SPC PWR fuel assemblies.

Section D-6.6.2 describes the HAC package array model for the IF-300 Cask containing GE BWR fuel assemblies. Section D-6.6.2.1 discusses the results of the HAC infinite array criticality calculations for the IF-300 Cask containing GE BWR fuel assemblies.

D-6.6.1 Arrays of IF-300 Casks with PWR Fuel under HAC

An infinite array of casks was modeled to conservatively represent 2N damaged packages. Once again, a triangular array configuration was used and the pitch was varied to determine the optimum distance between the casks (i.e., cask pitch). Sensitivity cases were run for various water densities within the containment (inleakage is assumed under HAC), various water densities for the interspersed moderation (between packages), and for different cask pitches. Figure D-6.6-1 illustrates the PWR Cask infinite array model under HAC. Note that, although the cask coolant jacket is not included in the model, because the model includes interspersed moderator between the infinite array of casks, the neutronic effects of the coolant jacket are accounted for.

D-6.6.1.1 Results of the HAC Infinite Array Package Evaluation for the IF-300 Cask With SPC PWR Fuel Assemblies from Robinson

Table D-6.6-1 summarizes the results of the calculations made for an infinite array of PWR casks which conservatively represents 2N damaged packages under HAC. The density of water both within the cask containment, and between (i.e., interspersed) the casks was varied in this evaluation, and the distance between the casks was also varied. Cases were run for the assemblies shifted radially inward and outward in the basket, as well as an their nominal centered locations. Because the effective multiplication of the system is maximized with the assemblies shifted outward, most cases were run with the assemblies shifted outward. The maximum k_{eff} including bias and uncertainties is 0.94830. This occurs for an infinite array of packages with the fuel assemblies shifted outward, with a distance of 10 cm between the casks, with water at a density of 1 g/cc in the containment, and with water at a density of 0.1 g/cc interspersed between the casks (case pso6i1-c). The results in Table D-6.6-1 indicate that the casks are effectively isolated from each other since the effective multiplication values for the infinite cask system are within the statistical uncertainties as the cask spacing is varied from 2 cm to 20 cm, and as the interspersed moderator density is varied from 0 g/cc to 1 g/cc. Example MCNP input files for the cases listed in Table D-6.6-1 are included in the Appendix.

It is concluded that an infinite array of IF-300 Casks will meet the criticality safety limit of k_{eff} < 0.95 when the cask contains six SPC PWR fuel assemblies with a maximum lattice average enrichment of 4.65 wt%

^{235}U loaded in the peripheral PWR basket locations (i.e., with the center basket location empty). Therefore, the IF-300 Cask meets the requirement that an array of 2 times N damaged packages stacked together in any arrangement with optimum interspersed hydrogenous moderation between the packages must be subcritical, where N in this case is equal to infinity. Note that the model did not use close full reflection of the array on all sides by water because an infinite array was analyzed.

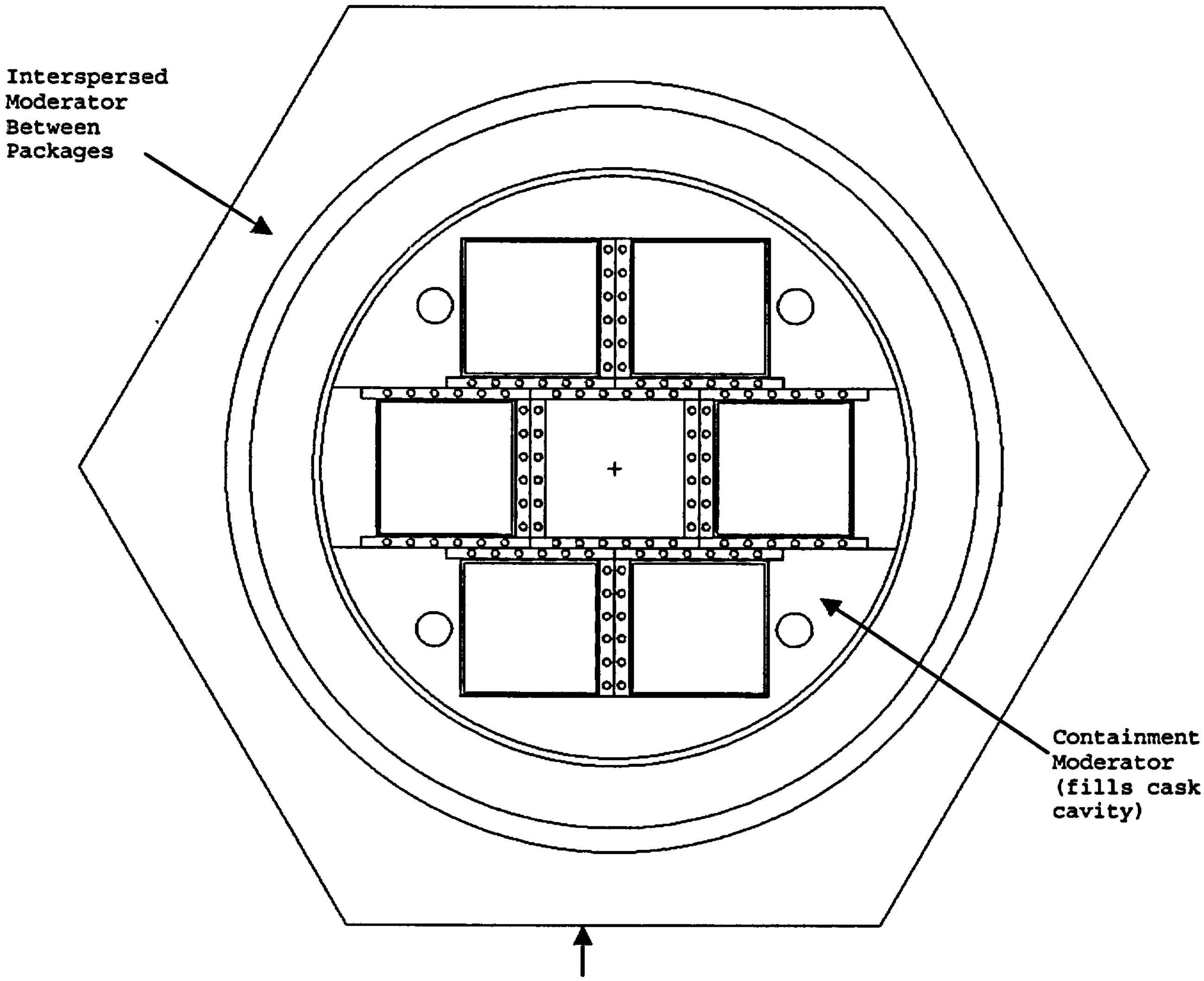

Figure D-6.6-1

PWR Cask HAC Infinite Array Model

Table D-6.6-1

Infinite Multiplication Factors for an Infinite Array of IF-300 Casks[a] Under HAC with 6 SPC PWR Fuel Assemblies with 4.65 wt% ^{235}U (2 sheets total)

MCNP input file[b]	Containment Water density g/cc	Inter-spersed Water density g/cc	Cask spacing cm	k_∞	σ	k_{tot}[c]
Pso6i11a	1	1	2	0.93310	0.00107	0.94782
Pso6i19a	1	0.9	2	0.93188	0.00098	0.94655
Pso6i17a	1	0.7	2	0.93144	0.00099	0.94612
Pso6i14a	1	0.4	2	0.92954	0.00096	0.94420
Pso6i1-a	1	0.1	2	0.93313	0.00108	0.94786
Ifp6i10a	1	0	2	0.93313	0.00108	0.94786
Psi6i10a	1	0	2	0.92643	0.00111	0.94118
Pso6i10a	1	0	2	0.93204	0.00101	0.94673
Pso6i11b	1	1	6	0.93134	0.00097	0.94601
Ifp6i19b	1	0.9	6	0.92552	0.00099	0.94020
Psi6i19b	1	0.9	6	0.92478	0.00098	0.93945
Pso6i19b	1	0.9	6	0.93112	0.00087	0.94574
Pso6i17b	1	0.7	6	0.93066	0.00100	0.94534
Pso6i14b	1	0.4	6	0.93168	0.00095	0.94634
Pso6i1-b	1	0.1	6	0.93353	0.00093	0.94818
pso6i10b	1	0	6	0.93273	0.00097	0.94740
pso6i11c	1	1	10	0.93001	0.00091	0.94465
pso6i19c	1	0.9	10	0.93139	0.00099	0.94607
pso6i17c	1	0.7	10	0.92901	0.00091	0.94365
ifp6i14c	1	0.4	10	0.92804	0.00099	0.94272
psi6i14c	1	0.4	10	0.92379	0.00092	0.93843
pso6i14c	1	0.4	10	0.92902	0.00100	0.94370
pso6i1-c	1	0.1	10	0.93360	0.00103	0.94830

MCNP input file[b]	Containment Water density g/cc	Inter-spersed Water density g/cc	Cask spacing cm	k_∞	σ	k_{tot}[c]
pso6i10c	1	0	10	0.93259	0.00098	0.94726
pso6i11d	1	1	16	0.93116	0.00101	0.94585
pso6i19d	1	0.9	16	0.93092	0.00095	0.94558
pso6i17d	1	0.7	16	0.92954	0.00093	0.94419
pso6i14d	1	0.4	16	0.93301	0.00097	0.94768
ifp6i1-d	1	0.1	16	0.92739	0.00100	0.94207
psi6i1-d	1	0.1	16	0.92751	0.00084	0.94211
pso6i1-d	1	0.1	16	0.93238	0.00096	0.94704
pso6i10d	1	0	16	0.93016	0.00107	0.94488
ifp6i1le	1	1	20	0.92889	0.00099	0.94357
psi6i1le	1	1	20	0.92513	0.00105	0.93984
pso6i1le	1	1	20	0.92880	0.00105	0.94351
pso6i19e	1	0.9	20	0.92948	0.00100	0.94416
pso6i17e	1	0.7	20	0.93128	0.00095	0.94594
pso6i14e	1	0.4	20	0.93101	0.00093	0.94566
pso6i1-e	1	0.1	20	0.93172	0.00100	0.94640
pso6i10e	1	0.0	20	0.93243	0.00099	0.94711
pso6i91c	0.9	1	10	0.89936	0.00156	0.91443
pso6i71c	0.7	1	10	0.81956	0.00169	0.83474
pso6i41c	0.4	1	10	0.67943	0.00157	0.69451
pso6i01c	0	1	10	0.33511	0.00122	0.34993

[a] Credit is only taken for 75% of the ^{10}B poison present in the cask basket.

[b] All MCNP input files with a "pso" prefix have the fuel assemblies shifted outward in the basket to maximize the assembly-to-assembly spacing. All cases with a "psi" prefix have the fuel assemblies shifted inward in the basket to minimize the assembly-to-assembly spacing. Cases with an "ifp" prefix have the assemblies in their nominal (centered) positions.

[c] $k_{tot} = k_\infty$ including bias and uncertainties (see Section D-6.7).