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Civilian Radioactive Waste Management System
Management & Operating Contractor

1999 Design Basis Waste Input Report for Commercial Spent Nuclear Fuel

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December 1999

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CHANGE HISTORY

Revision Number	Interim Change Number	Effective Date	Description and Reason for Change
00	00	09/99	Initial Issue.
01	00	12/99	Incorporates DOE comments to Rev 00. Editorial changes made to clarify text. No changes were made to actual data provided in the report.

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ACRONYMS

Acronyms	
ALARA	As Low As is Reasonably Achievable
BWR	Boiling Water Reactor
CALVIN	CRWMS Analysis and Logistics Visually Interactive Model
CDB	Characteristics Data Base
CFR	Code of Federal Regulations
CRD	CRWMS Requirements Document
CRWMS	Civilian Radioactive Waste Management System
CSNF	Commercial Spent Nuclear Fuel
DBWI	Design Basis Waste Input
DBWS	Design Basis Waste Stream
DCS	Delivery Commitment Schedule
DOE	U.S. Department of Energy
DPC	Dual-Purpose Canister
FDS	Final Delivery Schedule
ISF	Interim Storage Facility
LWR	Light Water Reactor
M&O	Management and Operating Contractor
MGR	Monitored Geologic Repository
MOX	Mixed Oxide
MPC	Multi-Purpose Canister
MTHM	Metric Tons of Heavy Metal
MTU	Metric Tons of Uranium
NRC	U.S. Nuclear Regulatory Commission
OCRWM	Office of Civilian Radioactive Waste Management (DOE)
PA	Performance Assessment
PWR	Pressurized Water Reactor
QARD	Quality Assurance Requirements and Description
RSC	Regional Servicing Contractor

ACRONYMS (Continued)

SNF	Spent Nuclear Fuel
SPC	Single Purpose Cask
SR	Site Recommendation
SS	Stainless Steel
TBD	To Be Determined
TBV	To Be Verified
TRIGA	Training Reactor, Isotopes, General Atomic
TSLCC	Total System Life Cycle Cost
UCF	Uncanistered Fuel
WPO	Waste Package Organization

Abbreviations

GWd	GigaWatt days
MWd	MegaWatt days
Zirc	Zircaloy

1. INTRODUCTION AND SUMMARY

The purpose of this document is to provide waste quantity and sequencing information that serves as the design basis for commercial spent nuclear fuel (CSNF) arriving at the repository, and the information on the transportation systems that will be used to deliver this fuel. It is intended as input for waste package and repository design analyses needed to ensure that facilities are flexible enough to be capable of receiving, unloading, handling, and emplacing the amounts and types of CSNF expected for receipt under realistic bounding conditions. It must be recognized that within the bounding limits, there will be CSNF with characteristics different from those described in this document, that must still be accepted into the Civilian Radioactive Waste Management System (CRWMS).

The previous Design Basis Waste Stream (DBWS) Report, issued in September 1996 [*Design Basis Waste Stream for Interim Storage and Repository* (CRWMS M&O 1996a)], relied on assumptions that are no longer valid based on current program planning, such as the elimination of the interim storage facility (ISF). Other changes include the acceptance of 83,800 Metric Tons of Heavy Metal (MTHM), instead of 63,000 MTHM, and the difference in when fuel is pulled from dry storage for shipment to the repository. To more accurately model utility fuel selection, all fuel that meets transportation limits is pulled from pool storage before it is pulled from dry storage. In addition, projected changes in spent fuel characteristics (resulting from utilities' desires to decrease fuel cycle costs) require examination to determine their effect on system design.

This analysis provides input useful for system throughput and sizing. However, it is based on assumptions of future system operations and forecasts of utility fuel selection that cannot be predicted with absolute certainty, as they rely on events outside of the control of the Office of Civilian Radioactive Waste Management (OCRWM). Consequently, there is no way of knowing the actual waste stream profile that will occur, and there are many uncertainties that affect the design of the CRWMS. Therefore, representative waste streams were developed as reasonable bounding cases based on consideration of the tradeoffs facing utilities and transportation contractors.

An Activity Evaluation has been conducted in accordance with QAP-2-0 REV 05, *Conduct of Activities*, and has determined that this report is not subject to the requirements of the Quality Assurance Requirements and Description (QARD) (DOE 1998b). Revision 00 of this report was prepared under PRO-TS-003 REV 01, *Development of Technical Documents Not Subject to QARD Requirements*. As this procedure has since been cancelled, REV 01 of this DBWI Report has been developed using AP-3.11Q, *Technical Reports*, as guidance.

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2. DISCUSSION

The development of any projected waste stream requires that some assumptions be made regarding system operations. The assumptions used in this analysis were developed using engineering judgement to project the CSNF system characteristics and events that are likely to occur. They were based on pragmatic requirements for system operations and do not represent extreme receipt scenarios that although possible, are judged to be highly improbable.

An area with a great deal of uncertainty is the order in which fuel will be designated for pickup at a given time. Within the limits of the contract and their annual allocation, Purchasers may choose any of their fuel for shipment or may even trade the allocation rights to another Purchaser. As acceptance time nears, a process will begin to determine exactly what fuel will be shipped to the repository. The Purchaser will submit a Delivery Commitment Schedule (DCS) 63 months prior to the acceptance year, which will allow a more precise prediction of fuel selection. However, until final trading by contract holders is completed and a detailed description of Purchaser's fuel is submitted 60 days prior to the scheduled transportation date, the specific fuel deliveries will not be known.

Since there is little certainty of discrete fuel selection until the delivery schedules are finalized, it was necessary to examine a variety of fuel selection methodologies to determine the impacts of changing the selection assumptions. Five cases were selected to provide the necessary variability. Out of the five cases, three were chosen as being sufficiently likely to warrant inclusion in this document. The other two cases were reviewed for comparison, but were considered either to be of low probability or bounded by other cases, and were therefore not included.

In the five cases that were reviewed, all of the assumptions remained the same with the exception of the primary fuel selection methodology. Two basic methods of fuel selection were employed. In the first method, the oldest fuel is accepted first with progressively younger fuel accepted next. In the second method, fuel of a certain age (5- and 10-years-old) is specified (provided that cask restrictions of the preferred cask are met) and progressively older fuel is chosen next. In addition, there is a variation of the second method in which all fuel is picked up in strict order of age with less dependence on available cask types. The five cases examined were:

- Case A - Fuel selection begins with 10-year-old fuel and progresses to older fuel.
- Case B - Fuel selection begins with 10-year-old fuel and progresses to older fuel in strict order of age.
- Case C - Fuel selection begins with oldest fuel still in pool and progresses to younger fuel.
- Case D - Fuel selection begins with 5-year-old fuel and progresses to older fuel in strict order of age (not included in this report).

- Case E - Fuel selection begins with 5-year-old fuel and progresses to older fuel (not included in this report).

All cases examined include the shipment of 5-year-old fuel since in each case younger fuel is selected if older fuel is not available. Cases D and E include large quantities of young, hot fuel (<10-years old). Under the terms of the Standard Contract, Purchasers can ship fuel as young as 5-years old as standard fuel (see Appendix F). Younger fuel can be shipped as non-standard fuel, as long as thermal limits are met. However, for such reasons as operational impacts and As Low As is Reasonably Achievable (ALARA) occupational exposure considerations, it was assumed that few Purchasers would choose to ship in such a stressing manner, reducing the probability of Cases D and E.

A more detailed description of these cases is provided in Appendix A.

2.1 GENERAL

The following general assumptions were made regarding system operations:

1. There will not be an ISF. All fuel will be shipped directly to the repository, starting in 2010.
2. Pool decommissioning and fuel transfer to dry storage will occur 5 years after reactor shutdown (except at the six sites identified in the following assumption).
3. Dual-purpose canisters (DPCs) will be used for required dry storage at most sites. All sites currently using single purpose casks (SPCs) for storage were evaluated in order to determine their likelihood of switching to DPCs in the future. The criteria used were: (1) number of SPCs currently loaded/ordered (the larger the number, the lower the probability), and (2) typical size of purchases (sites that purchased sufficient SPCs at one time for many years of storage were considered to have a lower probability of switching to DPCs). Based on these criteria, all sites except the following are assumed to continue using DPCs. For these sites, it is assumed that all SNF will be shipped to the MGR uncanistered.

   Calvert Cliffs

   North Anna

   Oconee

   Point Beach

   Surry

   Susquehanna

4. There are currently no loaded disposable multi-element, multi-purpose canisters (MPCs) in inventory, and projected quantities of MPCs are entirely predicated on a wide range of assumptions (e.g., no canisters because of cost, canistering only

those small-quantity assemblies identified as a separate handling geometry, or canistering all CSNF at certain utilities). This report assumes that no MPCs will be delivered to the Monitored Geologic Repository (MGR).

5. Single-element-size canisters are assumed to be dimensionally compatible with limits established for uncanistered assemblies and are assumed to be disposable.

6. The relatively small amount of civilian research Training Reactor, Isotopes, General Atomic (TRIGA) SNF is assumed to be standard fuel under the Standard Contract (See Appendix F.).

2.2 TRANSPORTATION

The following general assumptions were made regarding transportation:

1. Transportation of large capacity rail casks, by rail or heavy-haul truck, will be possible across Nevada.

2. The largest transportation cask that a reactor site can handle without major structural upgrades is assigned as the primary shipping cask for that site. If a system utilizing DPCs has been deployed at a utility, that system will likely be used for shipping some of the remaining utility inventory along with bare fuel capable transport systems.

3. Transportation cask assumptions reflect current licensing actions and plans, and current cask designs, including DPC systems being ordered or planned by the utilities. The resulting cask types are shown in Appendix A, Table A-1. For the most part, these casks are not actual vendor designs, but represent generic characteristics that are similar to existing cask designs. Engineering judgement was used to determine that casks with these characteristics could be developed.

4. The assumption that all rail shipments of waste to the repository will be by general freight with one cask per shipment was used in the development of these cases; however, the MGR should have the capability to accept shipments of up to five loaded rail cars at any point in time.

2.3 FUEL INVENTORY

A new methodology was used to revise fuel projections based on increased burnups observed at the utilities. As the existing fuel database did not account for the current industry trends toward higher burnups and enrichments (ANS 1997), it was determined that the database required an update. In order to accomplish this, a general method for estimating batch-average burnups of future SNF discharges by each reactor was developed. This method uses utility-provided projections for the first five discharge cycles and an extrapolation of the utility data to subsequent discharges. It was assumed that there would be no life extensions, no additional early shutdowns, and no new orders. This was felt to be a reasonable assumption, as potential early shutdowns and life extensions would tend to balance out, and the remaining small variations would have minimal effect on the analysis. Reactor service life was based on current license conditions and on plant life expectancy, as of June 1999 (DOE 1996).

The most aggressive plants plan to reach average burnups of 60 and 57 GigaWatt days/metric tons of uranium (GWd/MTU) for pressurized water reactors (PWRs) and boiling water reactors (BWRs) respectively, per assembly by 2015. Maximum burnups are expected to reach 75 and 65 GWd/MTU per assembly (PWR & BWR respectively). The burnup increase (excluding final discharge) over the original 1995 projection used in the Total System Life Cycle Cost (TSLCC) (DOE 1998a) is approximately 5.4 percent, with a corresponding 4.8 percent reduction in the amount of fuel (MTU) covered by the projection, for a new total of 83,800 MTHM versus 86,300 MTU (DOE 1998a). [*Calculation Method for the Projection of Future SNF Discharges* (Draft). A00000000-01717-0200-0052 REV 00, Vienna Virginia: CRWMS M&O. ACC: MOV.19990625.0001.] Table 1 provides a summary of the resulting fuel characteristics when this "Base Case" projection is added to the existing historical data. More detailed data can be found in Appendix B.

Table 1. Summary of CSNF Characteristics for Case B [a]

Fuel – Cladding Combination	Total MTU [c]	Number of Assemblies [c]	Burnup (MWd/MTU) [c]		Enrichment (Percent)	
			Minimum	Maximum	Minimum	Maximum
BWR-Zirc	29,600	166,600	200	65,600	0.70	4.28
PWR-Zirc	52,700	121,100	2,000	74,600	0.30	5.00
BWR-SS [b]	38	333	5,000	21,000	3.63	3.93
PWR-SS [b]	700	1,800	3,700	38,900	0.71	4.94
PWR-MOX [e]	800	1,800	17,000	69,000	4.30	4.30
Total BWR	29,600 [d]	166,900 [d]	200	65,600	0.70	4.28
Total PWR	54,200 [d]	124,800 [d]	2,000	74,600	0.30	5.00

NOTES: [a] Except where otherwise noted, reported totals include both existing and projected inventories.
[b] Actual historical numbers, as there are no projected discharges of this fuel type.
[c] Numbers, except where otherwise noted, are rounded to the nearest one hundred.
[d] Numbers may not add due to independent rounding.
[e] Projected.

An additional projection was also examined for its impact on the waste stream. This projection uses a higher maximum burnup with the same target date (2015) for reaching this value. The higher burnup is based on assuming an increase in the maximum enrichment from 5 percent to 5.5 percent. This resulted in peak burnups of over 80 GWd/MTU, and average burnups of 72 GWd/MTU for PWR fuel and 68 GWd/MTU for BWR fuel. The overall average burnup did not change significantly; however, the peak values increased dramatically. This projection is considered possible, though less likely to occur. Therefore, a detailed evaluation is not included in this study.

This report describes the incoming waste stream and does not levy system requirements. Consequently, the entire waste stream that is expected, based on the "Base Case" projection, is presented.

2.4 FUEL ACCEPTANCE

Allocation rights for CSNF will be assigned to Purchasers in accordance with the Acceptance Priority Ranking and Annual Capacity Report (DOE 1995a). The rate of acceptance is provided in Table 2.

Table 2. CSNF Assumed Annual Acceptance Rates

Year	Acceptance Rate (MTHM/year)
1999-2009	0
2010	400
2011	600
2012	1,200
2013	2,000
2014	3,000
2015-2039	3,000
2040	1,600
Total	83,800

SOURCE: Data Source (2010 – 2032):
CRWMS Requirements Document (CRD), Revision 5, Table 1 (DOE 1999).

The assumptions provided above were chosen to represent realistic bounding cases; however, the projected deliveries described in the following sections will differ from what will actually occur. While every attempt has been made to address all reasonable scenarios, the MGR must have sufficient flexibility to address deviations.

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3. RESULTS

CRWMS Analysis and Logistics Visually Interactive Model (CALVIN) Version 2.0 (CRWMS M&O 1999a) was used to generate the waste streams, by cask type, for this analysis. The fuel discharge database and the individual pool cask assignments were updated with the latest information, and the model was run using the assumptions described in Section 2.

The results are provided in two parts. Sections 3.1-3.3 provide data that is independent of the fuel selection process (physical characteristics, handling interfaces, and cask systems). Section 3.4 provides data that is affected by the fuel selection process. This includes throughput related data that describes variations in transportation cask arrivals (number and type) and quantity of assemblies by type. Whereas providing age and heat information implies a knowledge of fuel selection that does not currently exist, the large variation in the results provide for a stringent requirement that ensures a highly flexible system. The design data provided in Section 3.4 was selected because it provided the most limiting requirements from all of the cases reviewed while still being realistic.

This section provides an overview of the design basis waste input. More detailed design basis information is provided in the Appendices. In addition to the normal conditions presented, it is recognized that there will be off-normal conditions and events that need to be accommodated in the system design. These events, because they are uncommon, are discussed in Section B.1.4.

3.1 COMMERCIAL SPENT NUCLEAR FUEL PHYSICAL CHARACTERISTICS

Physical characteristics of the CSNF assemblies include length, cross-section, weight, and cladding, as reported by utilities through the RW-859 surveys or as provided by commercial vendors. In addition, they also include associated hardware and handling interfaces. Table 3 summarizes BWR assembly dimensions and weights by related groups, while Table 4 provides similar information for PWRs. Supporting detail, including figures, can be found in Appendix B.

Table 3. Bounding Design Basis Dimensions for BWR Assemblies [a]

Assembly Group	Design Basis Length (inches)	Design Basis Cross-Section (inches)	Primary Cladding	Design Basis Weight (pounds)	Percent of Total BWR Assemblies	Total BWR Assemblies
Big Rock Point	81.6 – 84.8	6.50 – 7.21	Zircaloy – 2	457 – 591	< 1	524
Misc. Shutdown Reactors [b]	95.0 – 141.4	4.00 – 6.31	Zircaloy – 2 [c]	276 – 480	1	1,615
GE BWR/2,3 and 4,5,6	171.0 – 178.0	5.24 – 6.07	Zircaloy – 2	556 – 725	99	164,800
All BWR Assemblies	81.6 – 178.0	4.00 – 7.21		276 – 725	100	166,939

NOTES: [a] Summarized from Tables B-1, B-3, and B-5 in Appendix B.
[b] Includes Dresden 1, Humboldt Bay, and La Crosse.
[c] LaCrosse cladding is 348H stainless steel.

Table 4. Bounding Design Basis Dimensions for PWR Assemblies [a]

Assembly Group	Design Basis Length (inches)	Design Basis Cross-Section (inches)	Primary Cladding	Design Basis Weight (pounds)	Percent of Total PWR Assemblies	Total PWR Assemblies
Misc. Shutdown Reactors [d]	111.7 – 140.2	6.27 – 8.60	304 Stainless Steel [f]	437 – 1612	2	2,460
Westinghouse, B&W, and others [b]	146.0 – 173.5	7.76 – 8.64	Zircaloy – 4	1096 – 1705	84	105,500
CE 16 x 16 and South Texas [c]	176.8–201.2 [e]	8.03 – 8.53	Zircaloy – 4	1430 – 1945	14	16,900
All PWR Assemblies	111.7 – 201.2	6.27 – 8.64		437 – 1945	100	124,860

NOTES: [a] Summarized from Tables B-2, B-4, and B-6 in Appendix B.
[b] Includes two B&W classes, CE 14 x 14, three Westinghouse classes, Fort Calhoun, Palisades, and St. Lucie 2.
[c] Includes CE 16 x 16, CE System 80, and South Texas.
[d] Includes Haddam Neck, Indian Pt. 1, San Onofre 1, Yankee Rowe.
[e] Without inserted control-rod assemblies, the CE 16 x 16 and CE System 80 assembly dimensions should not exceed 180.1 inches even with dimensional adjustments for irradiation growth, thermal expansion, and manufacturing tolerances.
[f] Yankee Rowe cladding is Zircaloy – 4.

3.1.1 Cladding

Cladding is predominantly but not exclusively Zircaloy (see Tables 3 and 4 for summaries and Tables B-7 and B-8 in Appendix B for details). Collectively, existing and projected discharged assemblies segregate as shown in Table 5.

Table 5. Cladding Summary By Fuel Type

CLADDING	PWR	BWR
Zircaloy	98.5%	99.8%
Stainless Steel	1.5%	0.2%

According to the Characteristics Data Base (CDB) (DOE 1987, Appendix 2A), BWR assemblies have Zircaloy-2 cladding, while PWR assemblies use Zircaloy-4. Reactors using assemblies with almost exclusively stainless steel (SS) cladding include the following:

- LaCrosse (348H stainless steel)
- Haddam Neck and San Onofre 1 (304 stainless steel)
- Indian Point 1 (304 stainless steel)

- Yankee Rowe has 76 assemblies (304 stainless steel) remaining, which did not get reprocessed following its transition to Zircaloy cladding.

Some assembly classes have a small number of assemblies (usually in the 1-10 total range) that are prototypes or lead test assemblies with cladding distinctly different from the others in that class (i.e., stainless steel cladding instead of Zircaloy). Changes away from zirconium-based cladding are not anticipated. However, ongoing trends toward higher burn-ups and increased cycle durations are leading to changes away from the use of Zircaloy-4 in PWRs in order to improve the corrosion characteristics of zirconium-based cladding. Since 1993, for example, Westinghouse Electric Corporation customers have been progressively increasing the use of zirconium-niobium alloy (ZIRLO) (1 percent Nb, 1 percent Sn, 0.1 percent Fe), such that now virtually all Westinghouse fabrication of domestic PWR fuel uses ZIRLO cladding (Telephone conversations with Mr. George Sabol, Westinghouse-Pittsburgh, and Mr. William Whitehead, Westinghouse-Columbia, September 1, 1999).

3.1.2 Assembly Handling Interfaces

Assembly handling differs between BWRs and PWRs. All BWR assemblies have a permanently attached U-shaped fixture at the top of the assembly, with assembly identification numbers located on the upper one-third of the U-shaped fixture (see Appendix B, Section B.1.3). The handling of PWR assemblies requires the use of grappling devices that generally grip the upper end fitting from the inside, with the added complication that the top of the assembly may contain a control-rod assembly, an absorber-rod assembly, or a plugging device (e.g., a thimble plug). These grappling devices are specific to the design and the fuel fabricator, and should be purchased from the fuel fabricator. There are eight groupings of PWR handling geometries (see Appendix B, Section B.1.3). Specific handling interfaces are to be verified (TBV). PWR assembly identification numbers are generally located on the face of the top nozzle or upper end fitting.

3.2 COMMERCIAL SPENT NUCLEAR FUEL REQUIRING CANISTERIZATION IN DISPOSABLE CANISTERS

There are 265 single-element-size canisters currently in inventory, subdivided into 145 containing SNF, 101 containing only nonfuel components, and 19 containing debris with a mixture of SNF and nonfuel components. At least 45 of these canisters fall outside the dimensional envelope that bounds all intact CSNF assemblies (minimum length of 81.6 inches, maximum length of 201.2 inches, minimum cross-section of 4 inches x 4 inches, and maximum cross-section of 8.64 inches).

The projected number of single-element-size canisters is shown below.

Table 6. Breakdown of Single-Element-Size Canisters

Number of Canisters	Type of Canisters
500 - 1,950 (assume 1,100)	Existing failed SNF that must be canistered prior to shipment
1,000	Projected failed assemblies (see Appendix C, Section C.1.2.2.2)
300	Material currently stored at utility sites in "baskets"
4,000	Non-fuel components (less if placed into larger canisters)
6,400	Total

Additional details on the content of the above canisters are provided in Appendix C.

As noted earlier, it has been assumed that single-element-size canisters will have cross-sectioned dimensions greater than or equal to 4 inches by 4 inches (or diameters ≥ 4 inches), and less than or equal to 9 inches by 9 inches (or diameters ≤ 9 inches). Thus, all of the canistered fuel has been treated effectively the same as uncanistered fuel, and is implicitly commingled with the uncanistered fuel as opposed to disposed in waste packages dedicated to canistered SNF. It is noted that the canisters of nonfuel components do not have an MTU equivalent. Therefore, the nonfuel canisters do not currently have any position in the delivery queue. Additionally, it has not been determined if all of this material requires geologic disposal, nor how those requiring repository disposal will be packaged. The current projections do not include the pickup, transport, or disposal of these canisters of nonfuel components. In general, it is expected that this material will be picked up in normal transport casks, possibly with special baskets, after SNF has been removed.

3.3 CASK AND CARRIER SYSTEMS

Design basis cask/carriers systems include legal-weight truck casks and rail casks loaded either on railcars or heavy-haul trucks. Bounding cask physical characteristics for CSNF transportation cask types projected to be received at the MGR are summarized in Table 7.

Table 7. Design Basis Cask Exterior Bounding Dimensions

Cask Physical Characteristics		BWR		PWR	
		Truck [a]	Rail	Truck [a]	Rail
With trunnions, and impact limiters	Length (inches)	245	327	234	340
	Diameter (inches)	90	140 [d]	90	140
With trunnions, w/o impact limiters	Length (inches)	200	233	200	233
	Diameter (inches)	48	103	48	103

Table 7. Design Basis Cask Exterior Bounding Dimensions (Continued)

Cask Physical Characteristics		BWR Truck [a]	BWR Rail	PWR Truck [a]	PWR Rail
Without trunnions or impact limiters	Length (inches)	200	233	200	233
	Diameter (inches)	48	99	48	99
Weight (tons)		26.8	150 [b]	27.1	150 [b]
Average Assembly Heat (watts) [c]		235-1,100	235-2,400	600-2,500	235-6,000

NOTES: [a] Includes both the small (1 PWR / 2 BWR) cask and the large (4 PWR / 9 BWR) cask.
[b] Includes 20 percent design margin for maximums over maximum projected cask weights.
[c] Inverse correlation between cask average assembly thermal output and number of assemblies transported.
[d] TN-68 storage cask is docketed with the NRC for a 10 CFR Part 71 (Transportation) license, but this license has not yet been issued. TN-68 has a diameter (with trunnions and impact limiters) of 144 inches.

A table providing the dimensions of individual cask types is included in Appendix D, along with references to the documents used to develop the table. Bounding characteristics for cask carrier systems are provided in Table 8.

Table 8. Design Basis Cask Carrier System Bounding Physical Characteristics

Carrier Physical Specifications [a]	Rail Car	Heavy-Haul Truck	Legal-Weight Truck
Maximum Length (feet)	67.5	220 [b]	59.75
Maximum Width (inches)	140 [c]	168	96
Maximum Height (inches)	181	168	162
Total Weight (pounds)	263,000 (4-axle car) 526,000 (8-axle car)	502,000	80,000
Maximum Per-Axle Gross Weight (pounds)	65,750	40,560 (TBV)	17,000

SOURCES: [a] Physical parameters taken from reference CRWMS M&O 1998b, except for heavy-haul truck maximum length.
[b] Parameter taken from reference PTG 1997.
[c] TN-68 will require a rail with a width of 144 inches (assuming it is licensed for transport).

3.4 CASK AND ASSEMBLY ARRIVAL DATA

Tables 9 through 14 provide a summary of the cask arrival results. Detailed data is provided in Appendix E. Table 9 provides the average and maximum cask arrivals grouped into five time periods (initial ramp up, ramp down, and 10-year increments in-between). Note that in a few specific cases, a mix of CSNF classes (e.g., at San Onofre) may be included in any cask or DPC shipment.

Table 9. Cask Shipments by Transportation Mode

Year	Annual Average [a]				Annual Maximum [a]			
	Truck	UCF Rail	DPC Rail	Combined [b]	Truck	UCF Rail	DPC Rail	Combined [b]
2010-2014	120	160	30	300	220	340	40	580
2015-2024	90	350	60	500	230	380	110	620
2025-2034	10	230	150	390	40	330	240	450
2035-2039	0	60	260	310	0	70	260	320
2040	0	40	140	180	0	40	140	180

NOTES: [a] Numbers have been rounded up to the nearest 10 casks.
[b] Combined is the maximum combined number of casks to arrive in any 1 year, not the sum of the maximum values shown in the table.

Table 10 displays the maximum quantity of assemblies expected to arrive within any given year, by type.

Table 10. Summary of Maximum Annual Assembly Arrivals

	BWR	PWR	Combined [a]
Maximum [b]	7,800	5,000	11,500

NOTES: [a] Combined is the maximum total number of assemblies arriving in any 1 year, not the sum of the maximum values shown in the table.
[b] Numbers have been rounded up to the nearest 100 assemblies.

While the total number of assemblies arriving did not change, there was a small variation in the number of assemblies arriving in a given cask type for the various cases; however, it was within the margin of uncertainty. The assembly totals are shown in Table 11.

Table 11. Total Assembly Arrivals by Cask Contents

Cask Contents	Assemblies [a]
Uncanistered SNF	156,400
DPCs	135,300

NOTE: [a] Numbers rounded to the nearest one hundred.

The Waste Package Operations and Performance Assessment Operations organizations require assembly characteristics information. As all CSNF is to be accepted, the enrichment and burnup characteristics provided in Table 1 apply for all cases. In addition, the age ranges were virtually identical regardless of the fuel selection method examined, with a minimum age of 5 years and a maximum age of 59–61 years. However, there was a considerable difference in the age distribution, as can be seen in Figures 1 and 2. These figures display assembly age for all fuel at arrival at the repository, based on the assumptions provided in Section 2, for Cases B and C.

Tables 12 and 13 show the heat output per assembly for BWR and PWR fuel, respectively. This reflects heat output for the entire waste stream at arrival at repository, based on the assumptions described in Section 2 for Cases A, B, and C. A more detailed breakout is provided in Appendix E for Case B, which was determined to be the most stressing case. The average heat per assembly is approximately 190 and 550 watts for BWR and PWR fuel, respectively. Note that the heats in these tables (and in Appendix E) were generated using the ORIGEN 2-based method. Table 14 provides an estimate of the heat distribution for the 72 TRIGA assemblies, assuming an age at arrival of 30 years. Note that the heat was based on Table 4.4.25 of Volume 2 of the CDB (DOE 1992a), scaled linearly with burnup and assembly/rod weight.

Table 12. Summary Heat Distribution for BWR Fuel (Watts) at Arrival [b]

Heat Range (Watts/Assembly)	Case A (percent)	Case B (percent)	Case C (percent)	Range [a] (percent)
0 - 49	5.6	5.6	5.2	5.2 - 5.6
50 - 99	22.8	24.6	18.4	18.4 - 24.6
100 - 149	15.6	16.0	21.0	15.6 - 21.0
150 - 199	11.8	10.9	14.2	10.9 - 14.2
200 - 249	16.3	8.9	18.1	8.9 - 18.1
250 - 299	9.2	10.5	13.8	9.2 - 13.8
300 - 349	10.6	13.3	5.8	5.8 - 13.3
350 - 399	6.2	8.1	2.3	2.3 - 8.1
400 - 449	1.2	1.4	0.9	0.9 - 1.4
450 - 499	0.4	0.5	0.3	0.3 - 0.5
500 - 549	0.2	0.2	0.0	0.0 - 0.2
550 - 599	0.1	0.0	0.0	0.0 - 0.1

NOTES: [a] Column may not add to 100 percent due to rounding.
[b] Based on Origen 2 heat code.

Table 13. Summary Heat Distribution for PWR Fuel (Watts) at Arrival [c]

Heat Range (Watts/Assembly)	Case A (percent)	Case B (percent)	Case C (percent)	Range [b] (percent)
0 - 99	1.0	1.0	0.9	0.9 – 1.0
100 - 199	8.1	8.3	7.0	7.0 - 8.3
200 - 299	16.2	17.9	13.9	13.9 - 17.9
300 - 399	13.9	14.8	16.3	13.9 - 16.3
400 - 499	9.7	8.8	13.8	8.8 - 13.8
500 - 599	9.0	7.9	13.4	7.9 - 13.4
500 - 699	14.4	7.6	12.7	7.6 - 14.4
600 - 799	9.4	8.6	8.3	8.3 - 9.4
800 - 999	12.3	14.3	8.1	8.1 - 14.3

Table 13. Summary Heat Distribution for PWR Fuel (Watts) at Arrival (Continued)

Heat Range (Watts/Assembly)	Case A (percent)	Case B (percent)	Case C (percent)	Range (percent)
1,000 – 1,199	2.5	6.5	2.3	2.3 - 6.5
1,200 – 1,399	1.0	2.0	0.9	0.9 – 2.0
1,400 – 1,599	0.6	0.5	0.5	0.5 - 0.6
1,600 – 1,799	0.3	0.2	0.2	0.2 - 0.3
1,800 – 1,999	0.1	0.1	0.1	0.1 - 0.1
2,100 – 2,199	0.0	0.0	0.0	0.0 – 0.0
MOX [a]	1.4	1.4	1.4	

NOTES: [a] Heat is not calculated for MOX fuel.
[b] Column may not add to 100 percent due to rounding.
[c] Based on Origen 2 heat code.

Table 14. Estimated Heat Distribution for TRIGA Fuel (Watts)

Heat Range (Watts/Assembly)	Percent of Assemblies [a]
0.00 - 0.09	39
0.10 - 0.19	6
0.20 - 0.29	1
0.30 - 0.39	1
0.40 - 0.49	6
0.50 - 0.59	13
0.60 - 0.69	7
0.70 - 0.79	4
1.10 - 1.19	1
1.20 - 1.29	4
1.40 - 1.49	3
1.50 - 1.59	4
1.60 - 1.69	4
1.70 - 1.79	3
1.90 - 1.99	1
2.20 - 2.29	1
2.30 - 2.39	1

NOTES; [a] Column may not add to 100 percent due to rounding.

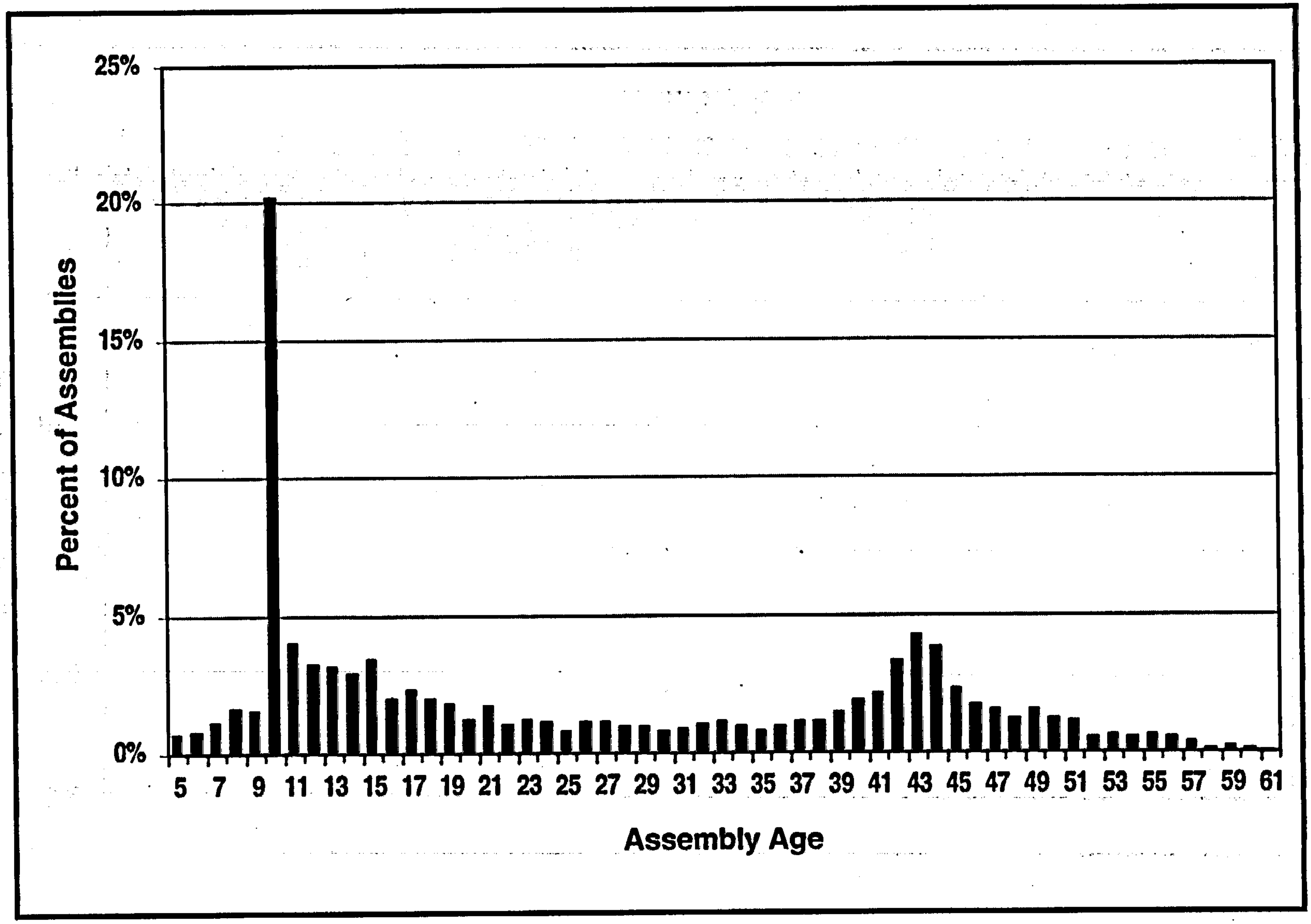

Figure 1. Age Distribution for Case B at Arrival

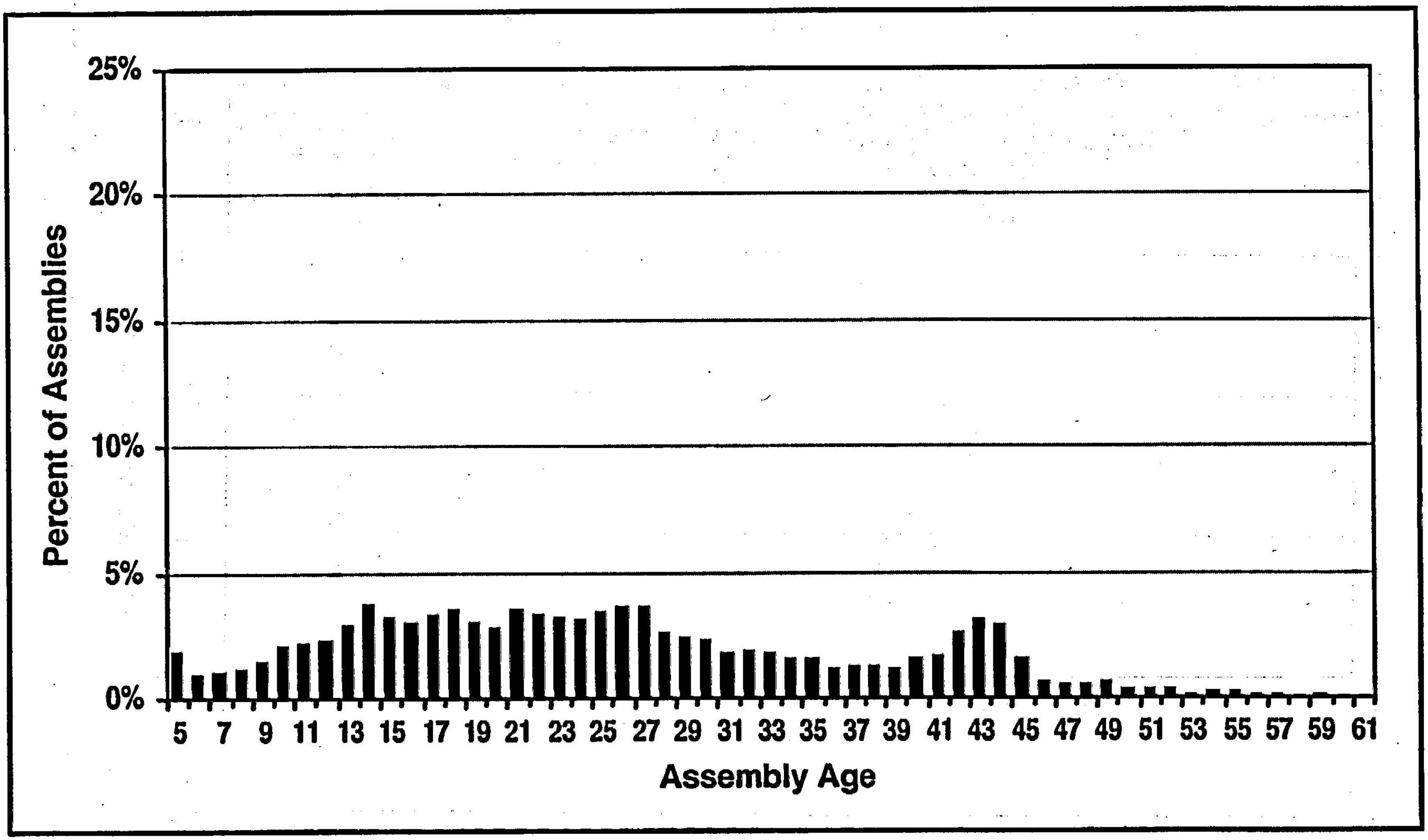

Figure 2. Age Distribution for Case C at Arrival

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4.2 STANDARDS, ORDERS, AND REGULATIONS

10 CFR (Code of Federal Regulations) 71. Energy: Packaging and Transportation of Radioactive Materials. Readily available.

10 CFR 961. Energy: Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste. Readily available.

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APPENDIX A

ADDITIONAL ASSUMPTIONS

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ADDITIONAL ASSUMPTIONS

A.1 CASK TYPE ASSUMPTIONS

As discussed in Section 2.2, current licensing actions and casks designs were examined to predict the most likely transportation casks. Table A-1 shows the characteristics of the resulting cask types, including capacity, weight, and heat limit. Table A-2 summarizes the 17 unique cask types and the 39 variations created by derating, switching BWR/PWR baskets, or removing the basket, to enable the shipment of DPCs.

Table A-1. Commercial Spent Nuclear Fuel Transportation Casks

Cask Designator	Description	Type	SPC/DPC	Capacity (Assys)	Nominal Hook Weight (Tons)	Heat Limit (Watts/ Assy)
Truck Casks						
P-T-1-SP	PWR NAC LWT	PWR	SPC	1	25	2,500
B-T-2-SP	BWR NAC LWT	BWR	SPC	2	25	1,100
P-T-4/4-SP	GA-4 LWT-4 Assemblies	PWR	SPC	4	26	617
P-T-4/3-SP	GA-4 LWT-3 Assemblies	PWR	SPC	3	26	740
P-T-4/2-SP	GA-4 LWT-2 Assemblies	PWR	SPC	2	26	1,234
B-T-9/9-SP	GA-9 LWT-9 Assemblies	BWR	SPC	9	26	235
B-T-9/7-SP	GA-9 LWT-7 Assemblies	BWR	SPC	7	26	303
B-T-9/5-SP	GA-9 LWT-5 Assemblies	BWR	SPC	5	26	406
B-T-9/4-SP	GA-9 LWT-4 Assemblies	BWR	SPC	4	26	530
B-T-9/2-SP	GA-9 LWT-2 Assemblies	BWR	SPC	2	26	730
SPC Rail Casks						
P-R-12-SP	PWR Generic Small SPC	PWR	SPC	12	75	1,000
B-R-32-SP	BWR Generic Small SPC	BWR	SPC	32	75	466
P-R-21-SP	PWR Generic Medium SPC	PWR	SPC	21	100	1,000
B-R-44-SP	BWR Generic Medium SPC	BWR	SPC	44	100	466
P-R-24-SP	PWR Generic Large SPC	PWR	SPC	24	125	706
B-R-68-SP	BWR Generic Large SPC	BWR	SPC	68	125	238
P-R-7-SP-HH	PWR Small SPC-HH	PWR	SPC	7	75	6,000
B-R-17-SP-HH	BWR Small SPC-HH	BWR	SPC	17	75	2,400
P-R-12-SP-HH	PWR Medium SPC-HH	PWR	SPC	12	100	6,000
B-R-32-SP-HH	BWR Medium SPC-HH	BWR	SPC	32	100	2,400
P-R-ST17-SP	South Texas SPC	PWR	SPC	17	125	1,000
P-R-ST7-SP-HH	South Texas SPC-HH	PWR	SPC	7	100	6,000
P-R-WV20-SP	PWR West Valley SPC	PWR	SPC	20	100	N/A

Table A-1. Commercial Spent Nuclear Fuel Transportation Casks (Continued)

Cask Designator	Description	Type	SPC/DPC	Capacity (Assys)	Nominal Hook Weight (Tons)	Heat Limit (Watts/ Assy)
P-R-9-SP-MOX	PWR MOX SPC	PWR	SPC	9	100	1,000
B-R-WV44-SP	BWR West Valley SPC	BWR	SPC	44	100	N/A
DPC Rail Casks						
P-R-24-OV	PWR Generic DPC	PWR	DPC	24	125	706
B-R-68-OV	BWR Generic DPC	BWR	DPC	68	125	238
P-R-WES21-OV	PWR WESFLEX DPC	PWR	DPC	21	100	1,000
B-R-WES44-OV	BWR WESFLEX DPC	BWR	DPC	44	100	466
P-R-VSC24-OV	Transtor DPC	PWR	DPC	24	125	1,000
P-R-MP24-OV	MP-187 DPC	PWR	DPC	24	125	764
P-R-HI24-OV	PWR HISTAR-100 DPC	PWR	DPC	24	125	706
B-R-HI68-OV	BWR HISTAR-100 DPC	BWR	DPC	68	125	238
P-R-NAC26-OV	PWR NAC UMS DPC	PWR	DPC	26	125	800
B-R-NAC56-OV	BWR NAC UMS DPC	BWR	DPC	56	125	300
P-R-YR36-OV	Yankee Rowe DPC	PWR	DPC	36	125	347
B-R-BP64-OV	Big Rock Pt DPC	BWR	DPC	64	125	378
P-R-ST17-OV	South Texas DPC	PWR	DPC	17	125	1,000
P-R-9-OV-MOX	PWR MOX DPC	PWR	DPC	9	100	1,000

Assy = Assembly
B = BWR
DPC = Dual-Purpose Canister
HH = High Heat
LWT = Legal Weight Truck
MOX = Mixed Oxide
N/A = Not Applicable
OV = Overpack (for DPCs)
P = PWR
R = Rail
SP = Single Purpose
SPC = Single Purpose Cask
T = Truck

Table A-2. Cask Type Summary

Cask Type	Description	BWR	PWR	SPC	DPC	Configurations
1	NAC LWT	X	X	2	--	2
2	GA-4	--	X	3 [a]	--	3
3	GA-9	X	--	5 [a]	--	5
4	Generic Small	X	X	2	--	2
5	Generic Medium/WESFLEX	X	X	3 [b]	3 [b]	6
6	Generic Large/HISTAR-100	X	X	2	4	6
7	Small HH	X	X	2	--	2
8	Medium HH	X	X	2	--	2
9	South Texas	--	X	1	1	2
10	South Texas HH	--	X	1	--	1
11	PWR West Valley	--	X	1	--	1
12	BWR West Valley	X	--	1	--	1
13	Transtor	--	X	--	1	1
14	MP-187	--	X	--	1	1
15	NAC UMS	X	X	--	2	2
16	Yankee Rowe	--	X	--	1	1
17	Big Rock Pt	X	--	--	1	1

NOTES: [a] Includes derated configurations.
[b] Includes PWR MOX.

A.2 SCENARIO DESCRIPTIONS

Case A: Fuel selection begins with 10-year-old fuel.

- Ten-year-old fuel was specified as the fuel to attempt to ship first from reactor pools. As long as this fuel does not exceed the design limits of the primary transportation cask, it will be shipped.

- Once all acceptable 10-year-old fuel has been shipped, the next oldest fuel will be tried, until the pool allocation has been filled or there is no more acceptable fuel greater than 10-years old at that site [allocation rights were assigned in accordance with Acceptance Priority Ranking and Annual Capacity Report, described in DOE 1995a]. At this point, younger fuel will be tried (from 9-years old through 5-years old).

- If the assemblies selected exceed the thermal limits for the primary cask, an alternate, more robust transportation cask (e.g., with a higher heat capacity) will be utilized to transport the assemblies.

- If the allocation still cannot be met (no acceptable 5-year or older fuel in the pool), fuel will be withdrawn from any onsite dry storage. If there is no onsite dry storage, the allocation will be deferred until next year.

This scenario allows the utility to eliminate large quantities of younger fuel, and still take ALARA occupational exposure considerations into account by loading the minimum number of casks.

Case B. This is a stricter variation of the Case A.

- A stricter variation of the Case A fuel selection method was also specified. Ten-year-old fuel was still specified as the fuel to attempt to ship first from reactor pools.

- If the limits of the primary cask are exceeded, rather than skipping that fuel, alternative, more robust casks are sequentially evaluated against assembly characteristics until an acceptable cask has been located.

- Once a cask type has been identified, the cask type is reset as the primary cask and the process repeats with the next fuel in the queue until the allocation is filled.

- Once all 10-year-old or older fuel has been accepted, younger fuel is selected (from 9-years old down to 5-years old).

- If the allocation still cannot be filled (no 5-year or older fuel in pool), fuel will be withdrawn from any onsite dry storage. If there is no onsite dry storage, the allocation will be deferred until next year.

This case assumes that the utilities are less concerned with the number of casks loaded than with the elimination of the hotter fuel from their pools. This case is considerably more stressing than Case A, since the requirement to take all fuel in strict age order results in the use of more transportation casks that can handle hotter fuel. These casks tend to have smaller capacities; therefore, more casks are required than with Case A. While Case B does not start with selection of the hottest fuel available (5-years old), it provides a relatively stressing, but still probable acceptance scenario.

Case C. Fuel selection begins with the oldest fuel still in the pool.

- Once the oldest fuel has been selected, or that fuel exceeds the primary cask limits, progressively younger fuel is tested for acceptance down to 5-year-old fuel.

- When there is no fuel that meets the primary cask limits, an alternative, more robust cask is tried.

- If the allocation still cannot be filled (no acceptable 5-year or older fuel in pool), then fuel will be withdrawn from any onsite dry storage. If there is no onsite dry storage, the allocation will be deferred until next year.

This case assumes that all utilities are willing to deliver the fuel that the DOE requests without regard to site-specific benefits of removing hotter fuel from utility pools. This case is included to provide sufficient variation in the possible waste to ensure that the design bounds plausible operational scenarios.

Case D. This is a stricter variation of Case B.

- The same strict variation of fuel selection was used as for Case B.
- The initial age of the fuel to begin attempting to ship was lowered to 5 years (from 10 years).
- Note that as the age of the fuel is already at the minimum, once all fuel older than the specified age is accepted from the pool, fuel is withdrawn from any onsite dry storage.

This case assumes that all utilities use the worst case fuel selection criteria allowed by the standard contract. This produces extremely stressing conditions for design, throughput, and thermal management. These worst case assumptions are counter to both CRWMS and utility site-specific considerations due to: (1) ALARA occupational exposure considerations for utility personnel, (2) the prohibitive amount of time required to load a large number of small casks, and (3) the large amount of lag storage required to cool fuel prior to emplacement in the repository. This case is theoretically possible, but not a practical limit. This case represents an extreme worst case scenario, and detailed analysis is not included in this report. Limits associated with this case should not be used as a design basis.

Case E. This is a variation of Case A.

- The same variation of fuel selection was used as for Case A.
- The initial age of the fuel to begin attempting to ship was lowered to 5 years (from 10 years).
- Note that as the age of the fuel is already at the minimum, once all acceptable fuel older than the specified age is accepted, fuel is withdrawn from any onsite dry storage.

This case is very similar to Case A and is considered to be of reasonable likelihood. While the number of transportation casks is roughly the same as Case A, the age distribution of the fuel is different. Due to the relatively low limits of the primary cask (most casks are designed to handle ten-year-old fuel), any fuel that would excessively stress the system is delayed. Because this case does not stress the system differently from Case A, and is bounded by Case B, it was not necessary to utilize this case for developing design data.

Table A-3 provides a summary of the cases. Note that all of the cases evaluated had at least 5 percent of the total acceptance being less than 10-years old (including some 5-year-old fuel).

Table A-3. Case Summary

Case	Initial Fuel Selection	Strict Age Order
A	10-year old	No
B	10-year old	Yes
C	Oldest still in pool	No
D	5-year old	Yes
E	5-year old	No

APPENDIX B

COMMERCIAL SPENT NUCLEAR FUEL PHYSICAL CHARACTERISTICS

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COMMERCIAL SPENT NUCLEAR FUEL PHYSICAL CHARACTERISTICS

B.1 SNF CHARACTERISTICS NOT EXPECTED TO VARY OVER TIME

B.1.1. Dimensions and Weights

Tables B-1 and B-2 provide an overview of the boundary physical characteristics for BWR and PWR fuel, respectively, used to develop the design basis dimensions provided in Section 3.1 (Tables 3 and 4). Tables B-3, B-5, B-7, and B-9 provide the logic used to develop bounding lengths, widths, cross sections, weights, and cladding materials, respectively for BWR fuel. Tables B-4, B-6, B-8 and B-10 provide the corresponding data for PWR fuel. Schematics for PWR and BWR assemblies are shown in Figures B-1 through B-12.

Table B-1. Summary of BWR Assembly Bounding Physical Characteristics by Class

Assembly Class	Design Basis Length (inches)	Design Basis Cross Section (inches)	Design Basis Weight (pounds)	Primary Cladding	Projected Number of Assemblies [b]	Percentage of Inventory (percent) [d]
Big Rock Point	81.6-84.6 [c]	7.21	591	Zircaloy-2	524	0.3
Dresden 1	136.4	5.20	384	Zircaloy-2 [a]	892	0.5
GE BWR/2,3	173.0	6.07	725	Zircaloy-2	34,700	20.8
GE BWR/4,5,6	178.0	6.07	705	Zircaloy-2	130,100	77.9
Humboldt Bay	96.0	5.21	324	Zircaloy-2 [a]	390	0.2
LaCrosse	103.6	6.31	480	348H Stainless Steel	333	0.2
				Total	166,939	100

NOTES: [a] Early stainless steel-clad assemblies were all reprocessed at West Valley.
[b] Actual values as used for known amounts, such as from shutdown reactors. Projected values are rounded to the nearest 100 assemblies.
[c] Lower number provides the minimum design basis length for BWR assemblies.
[d] Column may not add due to rounding.

Table B-2. Summary of PWR Assembly Bounding Physical Characteristics by Class

Assembly Class	Design Basis Length (inches)	Design Basis Cross Section (inches)	Design Basis Weight (pounds)	Primary Cladding	Projected Number of Assemblies [b]	Percentage of Inventory (percent) [c]
B&W 15x15	173.5	8.64	1,705	Zircaloy-4	11,400	9.1
B&W 17x17	173.5	8.64	1,679	Zircaloy-4	4	<0.1
CE 14x14	169.7	8.21	1,395	Zircaloy-4	8,900	7.1
CE 16x16	190.9	8.24	1,527	Zircaloy-4	7,100	5.7
CE System 80	195.0	8.27	1,455	Zircaloy-4	6,300	5.0
Fort Calhoun	158.6	8.22	1,312	Zircaloy-4	1,200	1.0
Haddam Neck	139.9	8.60	1,612	304 Stainless Steel	1,102	1.0

Table B-2. Summary of PWR Assembly Bounding Physical Characteristics by Class (Continued)

Assembly Class	Design Basis Length (inches)	Design Basis Cross Section (inches)	Design Basis Weight (pounds)	Primary Cladding	Projected Number of Assemblies [b]	Percentage of Inventory (percent)
Indian Point 1	140.2(TBV)	6.37 (TBV)	462(TBV)	304 Stainless Steel	160	0.1
Palisades	150.7	8.36	1,385	Zircaloy-4	1,500	1.2
San Onofre 1	139.9	8.15	1,379	304 Stainless Steel	665	0.5
Saint Lucie 2	170.7	8.24	1,391	Zircaloy-4	2,000	1.6
South Texas	201.2	8.53	1,945	Zircaloy-4 [a]	3,500	2.8
West. 14x14	166.4	7.87	1,457	Zircaloy-4 [a]	7,600	6.1
West. 15x15	166.9	8.54	1,662	Zircaloy-4 [a]	14,400	11.5
West. 17x17	168.9	8.54	1,687	Zircaloy-4 [a]	58,500	46.9
Yankee Rowe	113.0	7.72	822	Zircaloy-4	533	0.4
				Total	124,864	100.0

NOTES: [a] ZIRLO (1 percent Nb,1 percent Sn, 0.1 percent Fe) cladding progressively replaces Zircaloy-4 cladding, beginning about 1995; after 1999, almost 100 percent of Westinghouse SNF discharges are ZIRLO-clad.
[b] Actual values are used for known amounts, such as from shutdown reactors. Projected values are rounded to the nearest 100 assemblies.
[c] Column may not add due to rounding.

Table B-3. BWR Bounding Lengths [h]

Assembly Class	Reference Length [a] (inches)		Post-Irradiation Length [b,c] (inches)	Added Design Margin [d] (inches)	Design Basis Length (inches)
	Nominal	Variance			
Big Rock Point	84.0	+0 -2.0	82.0 – 84.8	-0.4 [f]	81.6 – 84.8 [g]
Dresden 1	134.4	+0.6 [i] -0	134.4 – 136.4	0	136.4
GE BWR/ 2,3	171.2	+0.1 -0.2 [i]	171.0 – 173.0	0	173.0
GE BWR/ 4,5,6	176.2	+0 -0.7	175.5 – 178.0	0	178.0
Humboldt Bay	95.0	+0 -0	95.0 – 96.0	-0.5 [f]	96.0
LaCrosse	102.5	+0 -0.1 [e]	102.4 – 103.6	0	103.6

NOTES: [a] Utility or manufacturer-supplied fabricated-fuel dimensions that may or may not include design tolerances; variances reflect differences among the various assembly types within the class.
[b] Lower end of range is the same as the shortest reference length. Upper end of range reflects a 1 percent expansion of the longest reference length, as described in CRWMS M&O 1996b, to cover assembly growth due to irradiation, thermal expansion, and deviations from nominal dimensions.

Table B-3. BWR Bounding Lengths [h] (Continued)

[c] Range reflects the adjusted shortest and longest assembly types in the class, with the positive variation rounded up to the nearest 0.1 inches, and the negative variation rounded down to the nearest 0.1 inches.
[d] Design margin in addition to that covered in Footnote b.
[e] Artifact of rounding convention used rather than a true variation within the assembly class.
[f] Assume 0.5 percent reduction in minimum length to conservatively cover manufacturing tolerances.
[g] Lower number provides the <u>minimum</u> design basis length for BWR assemblies.

SOURCES: [h] CDB (DOE 1992b) and EIA (DOE 1996) unless otherwise noted.
[i] CDB (DOE 1987, Appendix 2A), Volume 3.
[j] Drawing in the Physical Description Report section, rather than the Assembly Fuel Dimension Summary Table of CDB (DOE 1987, Appendix 2A), Volume 3.

Table B-4. PWR Bounding Lengths [h]

Assembly Class	Reference Length [a] (inches)		Post-Irradiation Length w/o Control Assembly [b, c] (inches)	Post-Irradiation Length with Control Assembly (inches)	Added Design Margin [d] (inches)	Design Basis Length (inches)
	Nominal	Variance				
B&W 15 x 15	165.7	+ 0 / - 0.1 [e]	165.6 – 167.4	173.5 [k]	0	173.5
B&W 17 x 17	165.7	+ 0.1 [e] / - 0	165.7 – 167.5	173.5 [k]	0	173.5
CE 14 x 14	157.0	+ 0.3 / - 0	157.0 – 158.9	169.7 [i]	0	169.7
CE 16 x 16	176.8	+ 0.1 [e] / - 0	176.8 – 178.7	190.9 [i]	0	190.9
CE System 80	178.3	+ 0 / - 0.1 [e]	178.2 – 180.1	195.0 [i]	0	195.0
Fort Calhoun	146.0	+ 2.9 [j] / - 0	146.0 – 150.4	158.6 [i]	0	158.6
Haddam Neck	137.1	+ 0.6 [j] / - 0.1	137.0 – 139.1	139.9 [i]	0	139.9
Indian Point 1	138.8	TBD	140.2 (TBV)	Same	0	140.2 (TBV)
Palisades	147.5	+ 1.7 [j] / - 0	147.5 – 150.7	Same	0	150.7
San Onofre 1	137.1	+ 0.6 [j] / -0.1	137.0 – 139.1	139.9 [i]	0	139.9
St. Lucie 2	158.2	+ 0 / - 0.1 [e]	158.1 – 159.8	170.7 [i]	0	170.7

Table B-4. PWR Bounding Lengths [h] (Continued)

Assembly Class	Reference Length [a] (Inches)		Post-Irradiation Length w/o Control Assembly [b, c] (Inches)	Post-Irradiation Length with Control Assembly (Inches)	Added Design Margin [d] (Inches)	Design Basis Length (Inches)
	Nominal	Variance				
South Texas	199.0 [f]	+0.2 [i] -0	199.0 – 201.2 [f]	Same	0	201.2
Westinghouse 14 x 14	159.8 [g]	+1.8 [j] -0.1	159.7 – 163.3	166.4 [i]	0	166.4
Westinghouse 15 x 15	159.8 [g]	+1.8 [j] -0.1	159.7 – 163.3	166.9 [k]	0	166.9
Westinghouse 17 x 17	159.8 [g]	+1.8 [j] -0.1	159.7 – 163.3	168.9 [i]	0	168.9
Yankee Rowe	111.8	+0 -0.1	111.7 – 113.0	Same	0	111.7 – 113.0

NOTES:
[a] Utility or manufacturer-supplied fabricated-fuel dimensions that may or may not include design tolerances; variances reflect differences among the various assembly types within the class.
[b] Lower end of range is the same as the shortest reference length. Upper end of range reflects a 1 percent expansion of the longest reference length, as described in (CRWMS M&O 1996b) to cover assembly growth due to irradiation, thermal expansion, and deviations from nominal dimensions.
[c] Range reflects the adjusted shortest and longest assembly types in the class, with the positive variation rounded up to the nearest 0.1 inches, and the negative variation rounded down to the nearest 0.1 inches.
[d] Design margin in addition to that covered in Footnote b.
[e] Artifact of rounding convention used rather than a true variation within the assembly class.
[f] Dimension is that with control rod fully inserted, as there were no dimensions provided in CDB (DOE 1987, Appendix 2A), Volume 3 for assembly without nonfuel components; based on other Westinghouse 17 x 17 assemblies, length without nonfuel components should be approximately 6 inches shorter.
[g] Dimension includes fully inserted upper-end plugging device (e.g., thimble plug) and a bottom-end nozzle, but excludes control rod assembly; removal of the upper-end plug will reduce assembly dimension by approximately 5 inches).

SOURCES:
[h] CDB (DOE 1992b) and EIA (DOE 1996) unless otherwise noted.
[i] CDB (DOE 1987, Appendix 2A), Volume 3.
[j] Drawing in Physical Description Report section, rather than the Assembly Fuel Dimension Summary Table of CDB (DOE 1987, Appendix 2A), Volume 3.
[k] CRWMS M&O 1996b.

Table B-5. BWR Bounding Cross-Sections (Widths)[g]

Assembly Class	Reference Cross-Section[a] (inches)		Post-Irradiation Assembly Cross-Section[b,c,d] (inches)	Added Design Margin[e] (inches)	Design Basis Cross-Section (inches)
	Nominal	Variance			
Big Rock Point	6.52	+ 0 - 0.02[h]	6.50 – 6.81[j]	+ 0.40	7.21
Dresden 1	4.28	+ 0.09[i] - 0.28[h]	4.00 – 4.80[j]	+ 0.40	5.20
GE BWR/ 2,3	5.44	+ 0.08[h] -0.20[h]	5.24 – 5.67	+ 0.40	6.07
GE BWR/ 4,5,6	5.44	+ 0.08[h] - 0.19[h]	5.24 – 5.67	+ 0.40	6.07
Humboldt Bay	4.67	+ 0 - 0.67[h]	4.00 – 4.81	+ 0.40	5.21
LaCrosse	5.62	+ 0 - 0.01[f]	5.61 – 5.91[j]	+ 0.40	6.31

NOTES: [a] Utility or manufacturer-supplied fabricated-fuel dimensions that may or may not include design tolerances; variances reflect differences among the various assembly types within the class.
[b] Includes fuel channels; per reference CRWMS M&O 1996b. No post-irradiation dimensional adjustments are required to cover assembly growth due to irradiation, thermal expansion, or deviations from nominal dimensions (within measurement error of the assembly), but 0.135 inches is added to address bowing and bulging (an additional 0.088 inches is added to the design margin for the 1-2 percent of channels used for more than one assembly lifetime, but this is not included in cross-sections in the table).
[c] Ranges address variation among various assembly classes, with the positive variation rounded up to the nearest 0.01 inches and the negative variation rounded down to the nearest 0.01 inches.
[d] Dimensions do not include 0.309 inches to cover spacer buttons and attachment clips that are located within the top 5 inches of the fuel channel.
[e] Design margin includes 0.309 inches to cover spacer buttons and attachment clips, plus 0.088 inches for bowing and bulging in channels used for more than one assembly lifetime. Maximum cross-sections include fuel channels, bow/bulge adjustment, and spacer buttons and attachment clips; however, all are excluded from minimum cross-sections.
[f] Artifact of rounding convention used rather than a true variation within the assembly class.

SOURCES: [g] CDB (DOE 1992b) and EIA (DOE 1996) unless otherwise noted.
[h] CDB (DOE 1987, Appendix 2A), Volume 3.
[i] Drawing in Physical Description Report section, rather than the Assembly Fuel Dimension Summary Table of CDB (DOE 1987, Appendix 2A), Volume 3.
[j] CRWMS M&O 1996b, Section 8.2.

Table B-6. PWR Bounding Cross-Sections (Widths) [f]

Assembly Class	Reference Cross-Section [a] (inches)		Post-Irradiation Assembly Cross-Section [b,c] (inches)	Added Design Margin [d] (inches)	Design Basis Cross-Section (inches)
	Nominal	Variance			
B&W 15 x 15	8.54	+ 0 - 0.01 [e]	8.53 – 8.54	+ 0.10	8.64
B&W 17 x 17	8.54	+ 0 - 0.01 [e]	8.53 – 8.54	+ 0.10	8.64
CE 14 x 14	8.10	+ 0.01 [h] - 0.00 [g]	8.10 – 8.11	+ 0.10	8.21
CE 16 x 16	8.10	+ 0.04 [h] - 0.00 [g]	8.10 – 8.14	+ 0.10	8.24
CE System 80	8.10	+ 0.07 [h] - 0.00 [g]	8.10 – 8.17	+ 0.10	8.27
Fort Calhoun	8.10	+ 0.02 [g] - 0.00	8.10 – 8.12	+ 0.10	8.22
Haddam Neck	8.42	+ 0.08 [g] - 0	8.42 – 8.50	+ 0.10	8.60
Indian Point 1	6.27	TBD	6.27 (TBV)	+ 0.10	6.37 (TBV)
Palisades	8.20	+ 0.06 [g] - 0	8.20 – 8.26	+ 0.10	8.36
San Onofre 1	7.76	+ 0.01 [e] - 0	7.76 – 8.05 [i]	+ 0.10	8.15
St. Lucie 2	8.10	+0.04 [g] - 0	8.10 – 8.14	+ 0.10	8.24
South Texas	8.43	+ 0 - 0	8.43	+ 0.10	8.53
Westinghouse 14 x 14	7.76	+ 0.01 [e] - 0	7.76 – 7.77	+ 0.10	7.87
Westinghouse 15 x 15	8.44	+ 0 - 0.02 [g]	8.42 – 8.44	+ 0.10	8.54
Westinghouse 17 x 17	8.44	+ 0 - 0.02 [g]	8.42 – 8.44	+ 0.10	8.54
Yankee Rowe	7.62	+ 0 - 0.05 [g]	7.57 – 7.62	+ 0.10	7.72

NOTES:
[a] Utility or manufacturer-supplied fabricated-fuel dimensions that may or may not include design tolerances; variances reflect differences among the various assembly types within the class.
[b] Per reference CRWMS M&O 1996b, no post-irradiation dimensional adjustments are required to cover bowing and bulging (greater assembly flexibility than in BWRs), or assembly growth due to irradiation, thermal expansion, or deviations from nominal dimensions (within measurement error of the assembly).
[c] Ranges address variation among various assembly classes, with the positive variation rounded up to the nearest 0.01 inches and the negative variation rounded down to the nearest 0.01 inches.
[d] Covers estimated manufacturing tolerances only (no bowing or bulging assumed per Footnote b).
[e] Artifact of rounding convention used rather than a true variation within the assembly class.

Table B-6. PWR Bounding Cross-Sections (Widths) [f] (Continued)

SOURCES: [f] CDB (DOE 1992b) and EIA (DOE 1996) unless otherwise noted.
[g] CDB (DOE 1987, Appendix 2A), Volume 3.
[h] Drawing in Physical Description Report section, rather than the Assembly Fuel Dimension Summary Table of CDB (DOE 1987, Appendix 2A), Volume 3.
[i] Per CDB (DOE 1987, Appendix 2A), protective cover used for dry storage and transport increases effective diameter to 8.043 inches.

Table B-7. BWR Bounding Weights [f]

Assembly Class	Assembly Weight [a, b, c] (pounds)	Nonfuel Component Weight [d] (pounds)	Required Design Margin [d] (pounds)	Design Basis Weight (pounds)
Big Rock Point	457 [g] – 465 [e]	101	+ 25	591
Dresden 1	329 [g, e]	30	+ 25	384
GE BWR/ 2,3	556 [g] – 620 [g, e]	80	+ 25	725
GE BWR/ 4,5,6	575 [g] – 600 [g, h, e]	80	+ 25	705
Humboldt Bay	276 [e]	23	+ 25	324
LaCrosse	376 [g] – 386 [e]	69	+ 25	480

NOTES: [a] May or may not include design tolerances or any miscellaneous material inserted into the assembly.
[b] Variation among various assembly classes, with the positive variation rounded up to the nearest 1 pound and the negative variation rounded down to the nearest 1 pound.
[c] Reported assembly weights may include some nonfuel components.
[d] No data on several key assembly types within the assembly class.
[e] All weights taken from CRWMS M&O 1996b, but could not be independently verified.

SOURCES: [f] As recommended in CRWMS M&O 1996b.
[g] CDB (DOE 1987, Appendix 2A), Volume 3.
[h] Manufacturer other than that used for the class name (e.g., Westinghouse 8 x 8 QUAD+ (EIA Assembly Type Code G4608W) for GE BWR/ 4, 5, 6.

Table B-8. PWR Bounding Weights [g]

Assembly Class	Assembly Weight [a, b, c] (pounds)	Nonfuel Component Weight [e] (pounds)	Required Design Margin [f] (pounds)	Design Basis Weight (pounds)
B&W 15 x 15	1,515	165	+ 25	1,705
B&W 17 x 17	1,505	149	+ 25	1,679
CE 14 x 14	1,270 [h] – 1,293 [h, i]	77	+ 25	1,395
CE 16 x 16	1,430	72	+ 25	1,527
CE System 80	1,430	N/A	+ 25	1,455
Fort Calhoun	1,220	67	+ 25	1,312
Haddam Neck	1,255 [h] – 1,421 [h, i]	166	+ 25	1,612

Table B-8. PWR Bounding Weights [g] (Continued)

Assembly Class	Assembly Weight [a, b, c] (pounds)	Nonfuel Component Weight [e] (pounds)	Required Design Margin [f] (pounds)	Design Basis Weight (pounds)
Indian Point 1	437 (TBV) [d]	N/A	+ 25	462 (TBV)
Palisades	1,338 [h] – 1,360	N/A	+ 25	1,385
San Onofre 1	1,233 [h] – 1,247	107	+ 25	1,379
St. Lucie 2	1,300	66	+ 25	1,391
South Texas	1,720	200	+ 25	1,945
Westinghouse 14 x 14	1,096 [h] – 1,302	130	+ 25	1,457
Westinghouse 15 x 15	1,432 [h] – 1,472	165	+ 25	1,662
Westinghouse 17 x 17	1,348 [h] – 1,482	180	+ 25	1,687
Yankee Rowe	720 [h] – 797	N/A	+ 25	822

NOTES: [a] May or may not include design tolerances or any miscellaneous material inserted into the assembly.
[b] Variation among various assembly classes, with the positive variation rounded up to the nearest 1 pound and the negative variation rounded down to the nearest 1 pound.
[c] Reported assembly weights may include some nonfuel components.
[d] No data on several key assembly types within the assembly class.

SOURCES: [e] All weights taken from CRWMS M&O 1996b but could not be independently verified.
[f] As recommended in CRWMS M&O 1996b.
[g] CRWMS M&O 1996b, unless otherwise noted.
[h] CDB (DOE 1987, Appendix 2A),Volume 3.
[i] Manufacturer other than that used for the class name (e.g., ANF 14 x 14 CE for the class CE 14 x 14, and Westinghouse 15 x15 [EIA Assembly Type Code XHN15W] for Haddam Neck).

B.1.2 Cladding and Other Materials

Table B-9. BWR Cladding Material Summary [c]

Assembly Class	Primary Cladding	Other Materials Used
Big Rock Point	Zircaloy-2	None [a]
Dresden 1	Zircaloy-2	None [b]
GE BWR/ 2,3	Zircaloy-2	None
GE BWR/ 4,5,6	Zircaloy-2	None
Humboldt Bay	Zircaloy-2	None [b]
LaCrosse	348H Stainless Steel	None

NOTES: [a] EIA (DOE 1996) reports all 12 x 12 assemblies were reprocessed at West Valley.
[b] All earlier stainless steel-clad fuel assemblies from Dresden 1 and Humboldt Bay were reprocessed at West Valley.

SOURCE: [c] CDB (DOE 1992b) and EIA (DOE 1996) unless otherwise noted.

Table B-10. PWR Cladding Material Summary [d]

Assembly Class	Primary Cladding	Other Materials Used
B&W 15 x 15	Zircaloy-4	None
B&W 17 x 17	Zircaloy-4	None
CE 14 x 14	Zircaloy-4	None
CE 16 x 16	Zircaloy-4	None
CE System 80	Zircaloy-4	None
Fort Calhoun	Zircaloy-4	None
Haddam Neck	304 Stainless Steel	Zircaloy [a]
Indian Point 1	304 Stainless Steel	None
Palisades	Zircaloy-4	None
San Onofre 1	304 Stainless Steel	None
St. Lucie 2	Zircaloy-4	None
South Texas	Zircaloy-4	None
Westinghouse 14 x 14	Zircaloy-4	ZIRLO [b]
Westinghouse 15 x 15	Zircaloy-4	ZIRLO [b]
Westinghouse 17 x 17	Zircaloy-4	ZIRLO [b]
Yankee Rowe	Zircaloy-4	348H Stainless St [c]

NOTES: [a] Four early test assemblies plus a fraction of the final discharge were Zircaloy-4-clad.
[b] ZIRLO (1 percent Nb,1 percent Sn, 0.1 percent Fe) cladding progressively replaces Zircaloy-4 cladding, beginning about 1995; after 1999, almost 100 percent of Westinghouse SNF discharges are ZIRLO-clad.
[c] Seventy-six assemblies of 17 x 18 Westinghouse-manufactured stainless steel-clad assemblies that were not reprocessed at West Valley will be delivered for repository disposal.

SOURCE: [d] CDB (DOE 1992b) and EIA (DOE 1996) unless otherwise noted.

B.1.3 Handling Interfaces

BWR and PWR assemblies handling interfaces are somewhat different. All BWR handling occurs via grappling of the permanently attached U-shaped fixture at the top of the assembly. Handling geometries are identical for GE BWR/ 2, 3 and GE BWR/ 4, 5, 6 assemblies, but the other reactor-specific classes are slightly different (usually a flatter U-shape fixture). See Figures B-1 through B-12. In all BWRs, assembly identification numbers are located on the upper one-third of the U-shaped fixture.

Handling of all PWR assemblies requires the use of assembly-type-specific remote grappling devices that generally grip the upper end fitting from the inside, with the added complication that the top of the assembly may contain a control-rod assembly, an absorber-rod assembly, or a plugging device (e.g., a thimble plug). These grappling devices are specific to the design and the fuel fabricator, and should be purchased from the fuel fabricator. Specific handling interfaces are TBV. There are eight general groupings of PWR handling geometries, as follows:

- B&W 15 x 15 and 17 x 17 (subtle differences between the two)
- CE 14 x 14 and 16 x 16 (including subtle differences between the two; between CE, ANF, and Westinghouse fabricated fuels for the 14 x 14 design; and for assemblies from Ft. Calhoun and St. Lucie 2)
- CE 16 x 16 System 80
- Palisades (including subtle differences between CE and ANF fabricated fuels)
- Yankee Rowe (including subtle differences between the four fuel fabricators)
- Westinghouse 14 x 14, 15 x 15, and 17 x 17 (subtle differences between the three and among fuel fabricators Westinghouse, ANF, and B&W)
- Haddam Neck and San Onofre 1 (subtle differences between the two and both include a variation of Westinghouse 15 x 15 design)
- South Texas (variation of Westinghouse 17 x 17).

PWR assembly identification numbers are generally located on the face of the top nozzle or upper end fitting. See Figures B-1 through B-12.

Where noted on assembly drawings (only about half had this information), assembly identification numbers are generally located on the face of the top nozzle or upper end fitting.

B.1.4 ANTICIPATED OFF-NORMAL PHYSICAL CONDITIONS

CSNF included under the definition of "off normal" contains non-BWR/PWR assemblies and assemblies undamaged when loaded into a transport cask, but found to be jammed or otherwise damaged at the time of unloading at the MGR. The only CSNF assemblies, which are not power reactor assemblies, are from commercial research reactors. This includes less than one MTU of TRIGA fuel currently stored at GA. This fuel ranges in enrichment from 20-93 percent. The assumption for this SNF shall be that it will be placed into DOE SNF standard canisters to make it physically interchangeable with DOE-owned TRIGA fuel. There may be other commercial research reactors whose fuel supply and disposal are not covered by an existing arrangement with DOE, but such situations have not been identified. The assumption for such fuel, although not currently identified, is that it will be placed into DOE SNF standard canisters that are appropriate for that fuel.

There are few recorded incidents of assemblies being damaged during transport or being jammed in cask baskets. It is unclear as to whether these incidents are exceedingly rare or simply represent no risk to public safety, and therefore have not been reported. The only recorded event involved a shipment where the cask inadvertently remained filled with water and there was considerable iron oxide in the water when the cask was opened. Given the reported frequency of problems with assemblies during transport, and the number of cask shipments to date relative to the annual shipments once the MGR is operational, the conservative assumption that should be used in design is that there will be one cask shipment per year, with at least one damaged assembly that was undamaged prior to transport. An additional conservative assumption for design is that biennially one cask shipment will have a problem that requires the cask be taken off-line and remediated on a case-by-case basis.

Badly bent or bowed assemblies are another design consideration. Given that the assembly must fit within a transport-cask basket to be transported to the MGR, the theoretical maximum bowed assembly is one that can fit within the largest cask basket (11 inches x 11 inches). Dimensional tolerances used in cross-section dimensions indicate bowing and bulging in BWRs is approximately 0.20 inches and is negligible in PWRs. However, there may be isolated instances of more significant bowing that must be addressed on a case-by-case basis. It can be assumed that the MGR will not receive a bent or bowed assembly with a maximum distortion beyond that accommodated by an 11-inch square basket.

B.2 PARAMETERS PROJECTED TO VARY OVER TIME

Parameters that are expected to vary over time include annual receipt rate, fuel age at time of receipt, initial enrichment, and fuel burn-up. Tables B-11 through B-15 show a breakdown of the SNF database described in Section 2. Each table is associated with a specific fuel type (BWR, PWR, BWR Stainless Steel, PWR Stainless Steel, and MOX). The tables show the annual discharges by amount (MTU and number of assemblies) and by minimum and maximum burnup and enrichment. Note that the bold line in these tables (between 1995 and 1996) indicates the end of historical data and the start of the projection. Assumption-specific (receipt rate and acceptance method) age and burnup results are provided in Appendix E.

Table B-11. Historical and Projected Annual Reactor Discharge Characteristics of BWR Fuel (excludes SS Fuel)

Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
1968	0.6	5	1,449	1,874	3.63	3.63
1969	9.8	96	5,240	29,000	1.47	3.63
1970	5.6	29	177	360	2.13	2.14
1971	64.7	413	858	22,591	1.83	3.63
1972	145.1	795	2,735	21,920	1.66	3.63
1973	87.5	514	972	25,527	1.83	3.63
1974	241.6	1,290	4,946	25,078	1.83	3.63
1975	222.8	1,198	3,709	24,588	1.83	2.51
1976	298.1	1,666	1,307	22,617	1.09	3.54
1977	379.3	2,015	5,111	25,312	1.09	3.54
1978	383.6	2,239	3,075	26,838	1.10	2.77
1979	396.5	2,103	9,338	30,638	1.10	3.71
1980	618.4	3,318	1,886	32,651	0.71	3.82
1981	458.7	2,467	8,963	37,050	2.12	2.89
1982	353.9	1,921	9,565	40,522	1.76	3.82
1983	479.4	2,624	12,824	34,208	2.19	3.82
1984	497.9	2,735	5,508	43,250	2.19	3.71
1985	539.9	2,962	2,642	42,428	0.71	3.31
1986	455.2	2,523	1,774	38,278	0.71	2.85
1987	589.2	3,244	3,000	43,000	0.70	3.43
1988	535.6	2,956	3,000	33,000	0.73	3.43
1989	697.1	3,827	3,000	36,000	0.71	3.43
1990	632.8	3,485	4,000	38,000	0.71	3.43
1991	588.0	3,260	3,000	36,750	0.71	3.40
1992	688.9	3,809	13,000	37,000	0.94	3.40
1993	699.7	3,883	15,000	39,000	1.64	3.43
1994	674.3	3,777	1,969	47,000	1.76	3.43
1995	788.8	4,445	9,860	43,299	1.63	3.48
1996	860.6	4,924	21,760	45,350	2.47	3.43
1997	649.2	3,736	6,715	46,795	1.50	3.45
1998	758.2	4,338	24,093	50,444	2.66	3.66
1999	629.0	3,608	28,850	53,485	2.99	3.83
2000	637.9	3,649	29,265	57,431	3.07	4.07
2001	714.8	4,092	29,220	58,581	3.09	4.11

Table B-11. Historical and Projected Annual Reactor Discharge Characteristics of BWR Fuel (excludes SS Fuel) (Continued)

Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
2002	617.8	3,539	28,657	59,358	3.05	4.11
2003	611.4	3,503	29,110	58,830	3.09	4.07
2004	734.5	4,192	26,104	57,186	2.76	4.01
2005	599.5	3,425	22,594	58,102	2.50	4.05
2006	707.0	4,073	12,335	60,637	1.90	4.14
2007	657.1	3,745	24,650	61,378	2.64	4.18
2008	586.4	3,350	30,595	61,856	3.14	4.19
2009	696.1	3,998	10,307	62,611	1.76	4.23
2010	701.9	4,010	11,723	63,100	1.86	4.24
2011	603.0	3,464	11,420	63,870	1.84	4.28
2012	878.1	5,045	10,290	64,368	1.76	4.21
2013	724.9	4,106	10,802	65,154	1.79	4.24
2016	593.7	3,369	11,229	65,550	1.82	4.28
2017	244.4	1,392	35,977	65,550	3.40	4.28
2018	332.0	1,890	10,352	65,550	1.76	4.28
2019	299.5	1,703	36,104	65,550	3.44	4.28
2020	252.0	1,442	37,067	65,550	3.47	4.28
2021	184.7	1,054	36,830	65,550	3.48	4.28
2022	638.5	3,639	10,606	65,550	1.78	4.28
2023	405.3	2,332	10,480	65,550	1.77	4.28
2024	370.8	2,127	12,342	59,847	1.90	4.18
2025	325.7	1,848	10,645	65,550	1.78	4.28
2026	529.8	2,998	9,798	47,989	1.72	3.56
2027	33.5	194	48,450	65,550	4.28	4.28
2028	0.0	0	0	0	0.00	0.00
2029	136.1	788	14,398	58,440	2.05	4.11
2030-2035	0.0	0	0	0	0.00	0.00

NOTES: The bold line in these tables indicates the end of historical data and the start of the projection. Assumption-specific (receipt rate and acceptance method) age and burnup results are provided in Appendix E.

Table B-12. Historical and Projected Annual Reactor Discharge Characteristics of PWR Fuel (excludes SS Fuel)

Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
1968	0.0	0	0	0	0.00	0.00
1969	0.0	0	0	0	0.00	0.00
1970	0.0	0	0	0	0.00	0.00
1971	4.6	12	5,856	8,653	3.41	3.47
1972	60.2	153	9,921	21,526	2.27	3.47
1973	25.6	57	13,420	25,643	1.85	3.26
1974	175.0	419	4,244	30,441	1.86	3.70
1975	281.2	691	2,769	30,185	1.39	3.70
1976	360.0	825	11,008	32,282	1.39	3.40
1977	445.1	1,054	15,155	38,991	1.85	3.97
1978	679.4	1,613	9,614	40,200	1.01	3.70
1979	698.7	1,608	8,910	40,007	1.86	3.31
1980	577.1	1,351	12,034	44,438	2.01	3.40
1981	654.1	1,532	11,930	42,764	1.86	3.70
1982	640.4	1,491	12,665	55,349	1.98	3.70
1983	751.1	1,727	7,409	50,598	1.86	3.46
1984	819.4	1,884	9,908	42,910	1.72	3.70
1985	841.8	1,984	13,758	47,932	0.71	3.70
1986	965.4	2,218	11,070	53,000	1.61	3.98
1987	1,087.1	2,540	14,000	44,000	1.61	3.98
1988	1,075.3	2,523	17,000	57,000	1.89	4.05
1989	1,187.5	2,718	10,000	58,000	1.03	4.04
1990	1,476.5	3,420	13,000	52,000	1.50	4.20
1991	1,231.1	2,816	9,000	52,000	1.60	4.20
1992	1,525.1	3,514	5,000	54,000	1.60	4.21
1993	1,481.9	3,417	12,000	54,000	1.21	4.40
1994	1,210.5	2,829	2,000	54,750	0.30	4.45
1995	1,674.3	3,849	11,821	56,175	2.10	4.45
1996	1,571.0	3,649	11,773	54,484	1.73	4.38
1997	1,604.7	3,641	12,325	58,614	1.77	4.73
1998	947.9	2,177	31,344	62,243	3.47	4.66
1999	1,470.6	3,377	28,050	61,545	3.12	4.62
2000	1,171.8	2,677	33,823	61,914	3.60	4.64
2001	1,260.1	2,881	33,960	62,100	3.54	4.71

Table B-12. Historical and Projected Annual Reactor Discharge Characteristics of PWR Fuel (excludes SS Fuel) (Continued)

Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
2002	1,302.8	3,001	34,850	61,308	3.54	4.69
2003	1,224.7	2,793	33,960	62,914	3.70	4.75
2004	1,124.2	2,563	34,965	63,339	3.65	4.68
2005	1,258.6	2,896	35,994	64,588	3.66	4.76
2006	1,129.9	2,589	35,668	64,778	3.79	4.77
2007	1,192.8	2,727	11,888	67,691	1.74	4.98
2008	1,129.1	2,600	37,084	66,080	3.74	4.83
2009	1,169.4	2,684	15,000	69,000	1.99	5.00
2010	1,051.1	2,414	14,023	67,408	1.91	4.91
2011	1,117.2	2,579	36,997	69,000	3.80	5.00
2012	986.5	2,227	14,091	68,588	1.92	4.89
2013	1,581.5	3,654	12,977	71,413	1.83	5.00
2014	1,224.6	2,792	12,497	69,285	1.79	4.98
2015	841.3	1,934	14,811	71,305	1.98	5.00
2016	948.5	2,154	13,685	69,441	1.89	4.84
2017	911.3	2,081	16,567	70,666	2.12	5.00
2018	545.4	1,231	12,609	69,000	1.80	4.89
2019	542.6	1,226	42,178	73,166	4.15	5.00
2020	688.3	1,543	14,438	70,925	1.95	4.91
2021	773.5	1,758	11,146	71,857	1.68	5.00
2022	364.9	824	15,917	69,912	2.07	4.94
2023	613.6	1,419	13,229	74,612	1.85	5.00
2024	651.2	1,508	14,987	72,069	1.99	4.96
2025	619.3	1,383	15,230	73,152	2.01	5.00
2026	588.4	1,330	15,059	71,592	2.00	5.00
2027	503.1	1,105	15,489	69,000	2.03	4.93
2028	159.0	317	15,693	69,000	2.05	4.91
2029	134.2	313	16780	69,000	2.14	4.91
2030	123.8	296	16,780	69,000	2.14	4.91
2031	0.0	0	0	0	0.00	0.00
2032	35.6	81	51,000	69,000	4.52	4.63
2033	100.6	237	16,648	69,000	2.13	4.88

Table B-12. Historical and Projected Annual Reactor Discharge Characteristics of PWR Fuel (excludes SS Fuel) (Continued)

Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
2034	0.0	0	0	0	0.00	0.00
2035	98.0	213	16,581	67,299	2.12	4.86

NOTES: The bold line in these tables indicates the end of historical data and the start of the projection. Assumption-specific (receipt rate and acceptance method) age and burnup results are provided in Appendix E.

Table B-13. Annual Reactor Discharge Characteristics of BWR SS Fuel

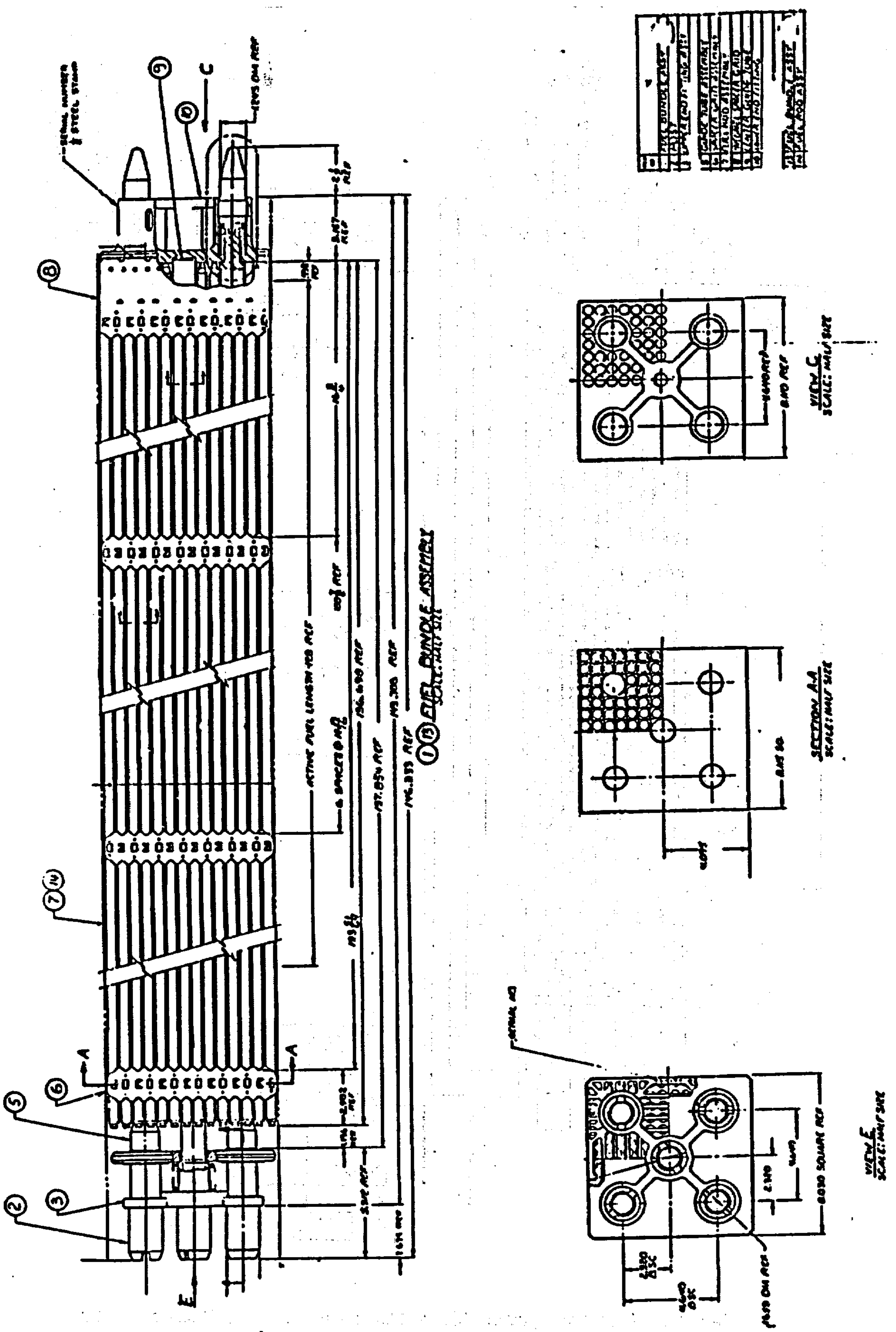
Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
1968	0.0	0	0	0	0.00	0.00
1969	0.0	0	0	0	0.00	0.00
1970	0.0	0	0	0	0.00	0.00
1971	0.0	0	0	0	0.00	0.00
1972	0.7	6	11,185	12,136	3.63	3.63
1973	6.0	50	8,766	16,736	3.63	3.63
1974	0.0	0	0	0	0.00	0.00
1975	3.0	25	7,575	20,988	3.63	3.92
1976	0.0	0	0	0	0.00	0.00
1977	3.9	32	7,609	19,224	3.92	3.93
1978	0.0	0	0	0	0.00	0.00
1979	3.4	28	12,502	14,679	3.92	3.93
1980	1.4	12	15,728	16,673	3.92	3.93
1981	0.0	0	0	0	0.00	0.00
1982	3.3	30	11,006	16,731	3.63	3.69
1983	2.4	22	15,776	18,662	3.69	3.69
1984	0.0	0	0	0	0.00	0.00
1985	3.0	28	16,341	19,824	3.69	3.70
1986	3.0	28	16,279	19,341	3.68	3.70
1987	7.8	72	5,000	18,000	3.68	3.71
1988-2035	0.0	0	0	0	0.00	0.00

Table B-14. Annual Reactor Discharge Characteristics of PWR SS Fuel

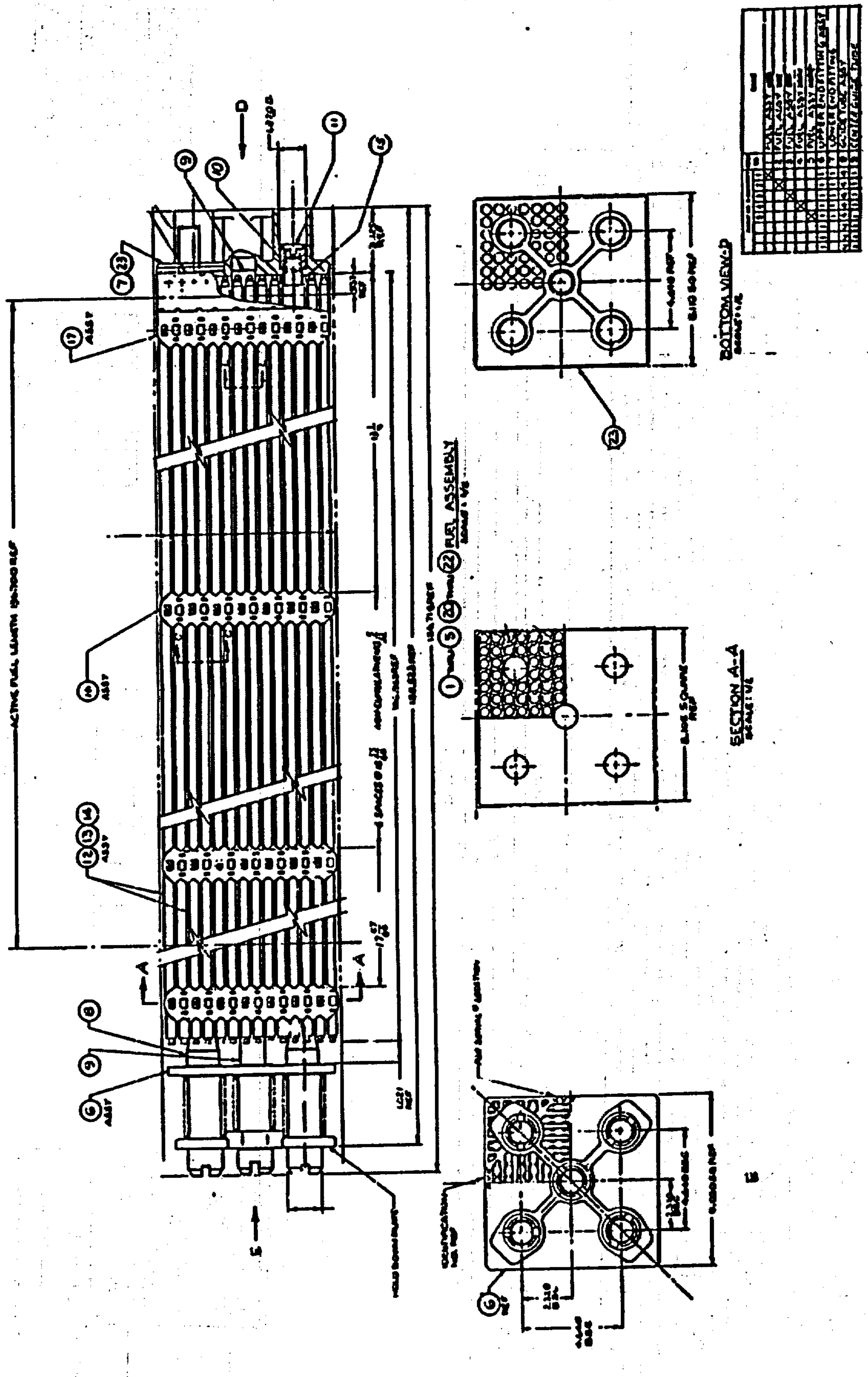
Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
1968	0.0	0	0	0	0.00	0.00
1969	0.0	0	0	0	0.00	0.00
1970	39.0	99	10,925	19,413	3.02	3.67
1971	39.9	101	22,375	26,593	3.02	3.40
1972	39.7	129	19,054	34,275	3.23	4.94
1973	41.4	108	19,233	32,658	0.71	3.87
1974	32.7	156	3,713	30,482	2.83	4.94
1975	40.5	106	20,790	34,844	3.16	4.94
1976	41.0	106	28,481	33,070	3.40	4.00
1977	21.8	53	31,892	36,713	3.65	4.01
1978	19.2	52	29,755	34,573	3.40	4.00
1979	20.2	49	30,052	35,936	3.98	4.01
1980	41.0	105	23,556	38,934	3.99	4.01
1981	21.8	53	29,933	34,957	4.00	4.00
1982	0.0	0	0	0	0.00	0.00
1983	20.2	49	31,942	35,899	4.00	4.00
1984	21.8	53	33,889	37,519	3.99	4.00
1985	19.3	52	29,579	38,001	3.99	4.00
1986	23.0	56	8,429	36,702	4.00	4.01
1987	21.8	53	32,000	37,000	4.00	4.00
1988	19.2	52	31,000	35,000	3.98	3.99
1989	20.6	50	11,000	34,000	4.00	4.00
1990	14.8	40	30,000	30,000	3.98	4.00
1991	21.8	53	34,000	38,000	4.00	4.00
1992	58.1	157	10,000	35,000	4.00	4.00
1993	21.9	53	23,000	37,000	3.00	4.00
1994	0.0	0	0	0	0.00	0.00
1995	21.8	53	30,963	37,574	3.99	4.00
1996	1.7	4	32,500	32,500	4.00	4.00
1997-2035	0.0	0	0	0	0.00	0.00

Table B-15. Historical and Projected Annual Reactor Discharge Characteristics of PWR MOX Fuel

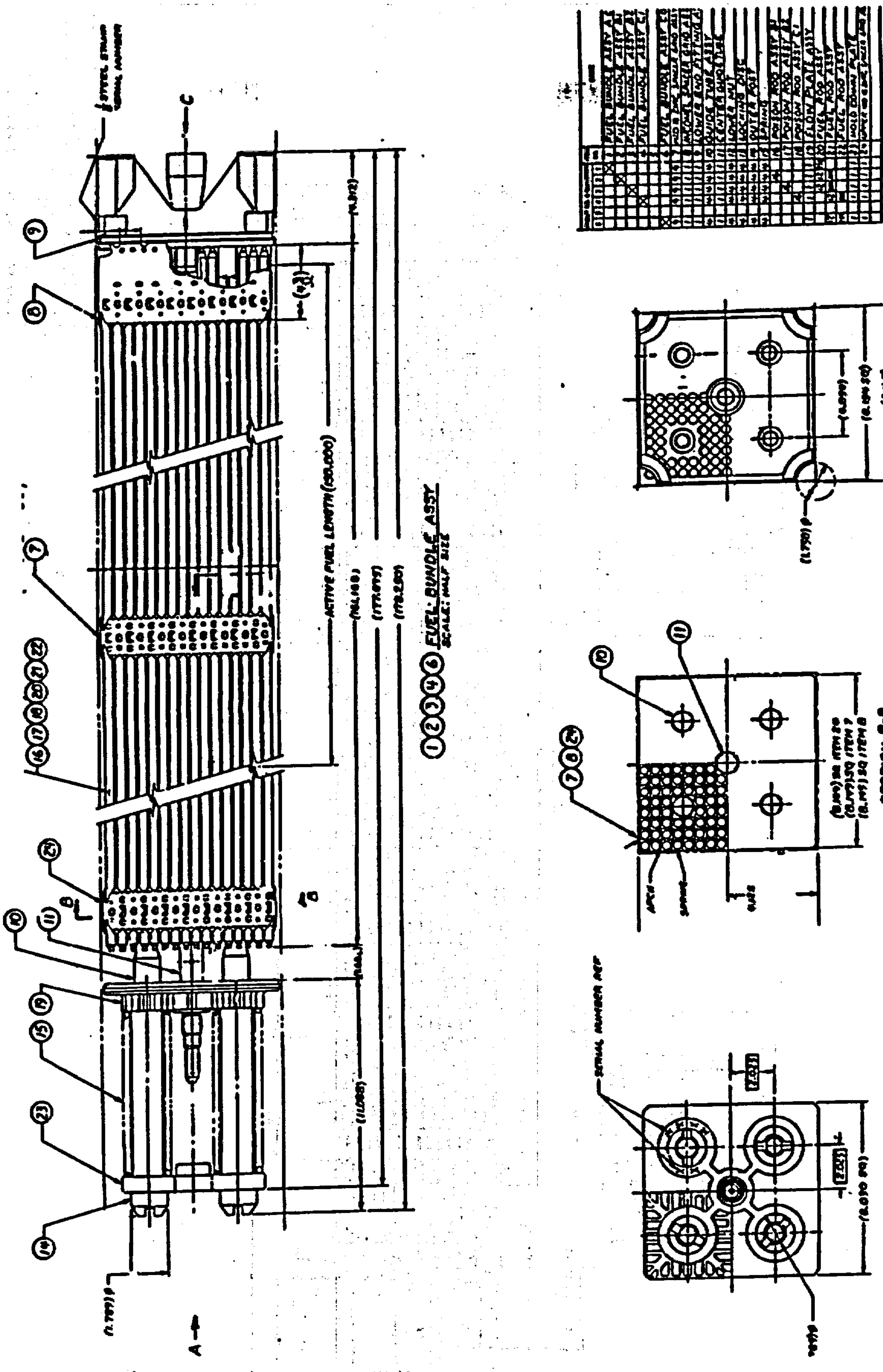
Year	Total MTU	Number of Assemblies	Burnup (MWd/MTU)		Enrichment (percent)	
			Minimum	Maximum	Minimum	Maximum
1968-2009	0.0	0	0	0	0.00	0.00
2010	55.1	120	40,800	55,200	4.30	4.30
2011	54.9	120	40,800	55,200	4.30	4.30
2012	55.4	121	40,800	55,200	4.30	4.30
2013	53.7	117	40,800	55,200	4.30	4.30
2014	53.1	116	40,800	55,200	4.30	4.30
2015	54.0	118	40,800	55,200	4.30	4.30
2016	52.8	115	40,800	55,200	4.30	4.30
2017	51.7	113	40,800	55,200	4.30	4.30
2018	75.3	164	17,000	69,000	4.30	4.30
2019	39.4	86	40,800	55,200	4.30	4.30
2020	68.0	148	17,000	69,000	4.30	4.30
2021	81.2	178	17,000	69,000	4.30	4.30
2022	13.2	29	40,800	55,200	4.30	4.30
2023	56.1	123	17,000	69,000	4.30	4.30
2024	56.6	124	17,000	69,000	4.30	4.30
2025	3.6	8	40,800	55,200	4.30	4.30
2026-2035	0.0	0	0	0	0.00	0.00



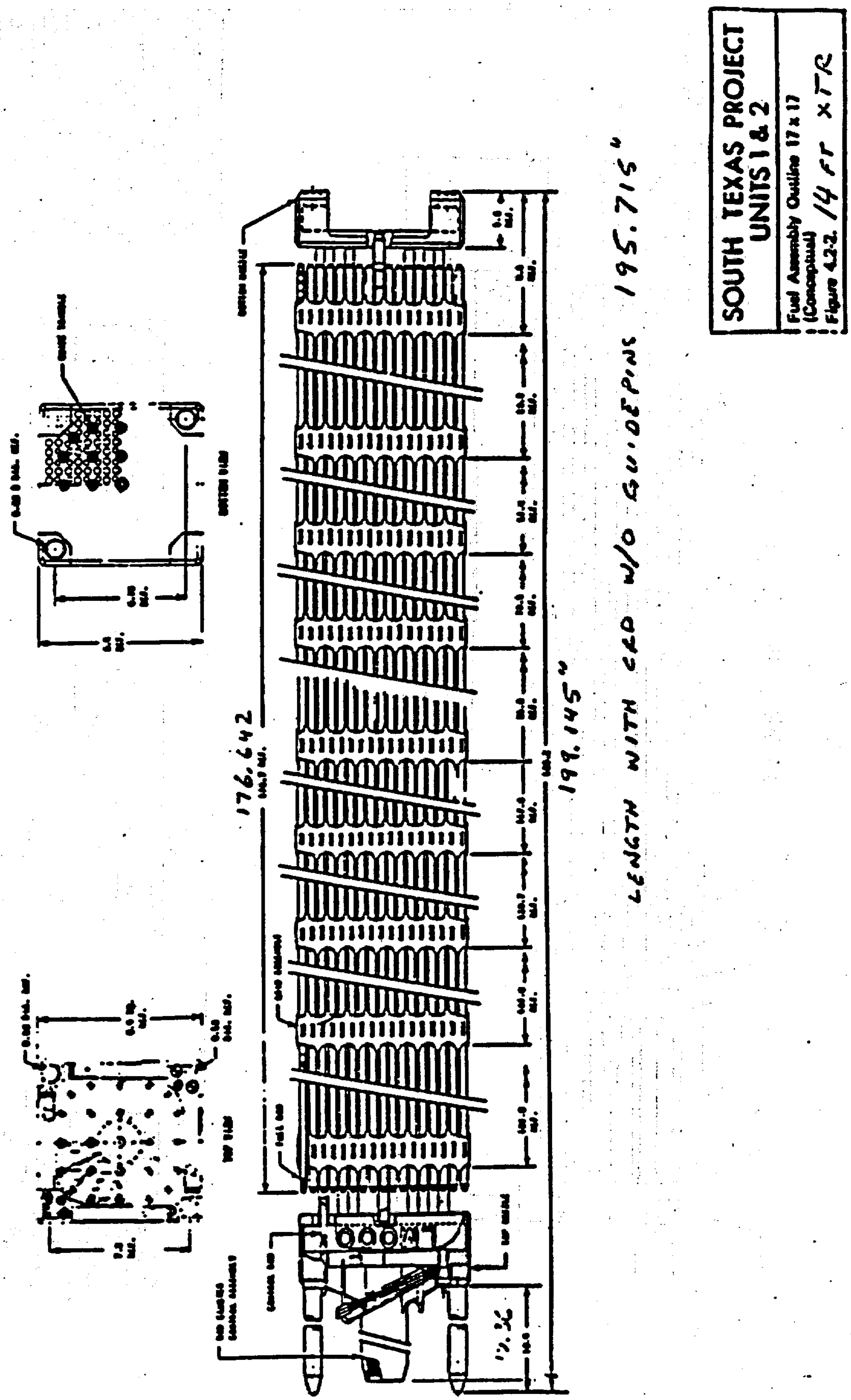
B-1. Fuel Assembly Drawing (Ft. Calhoun Station 1)



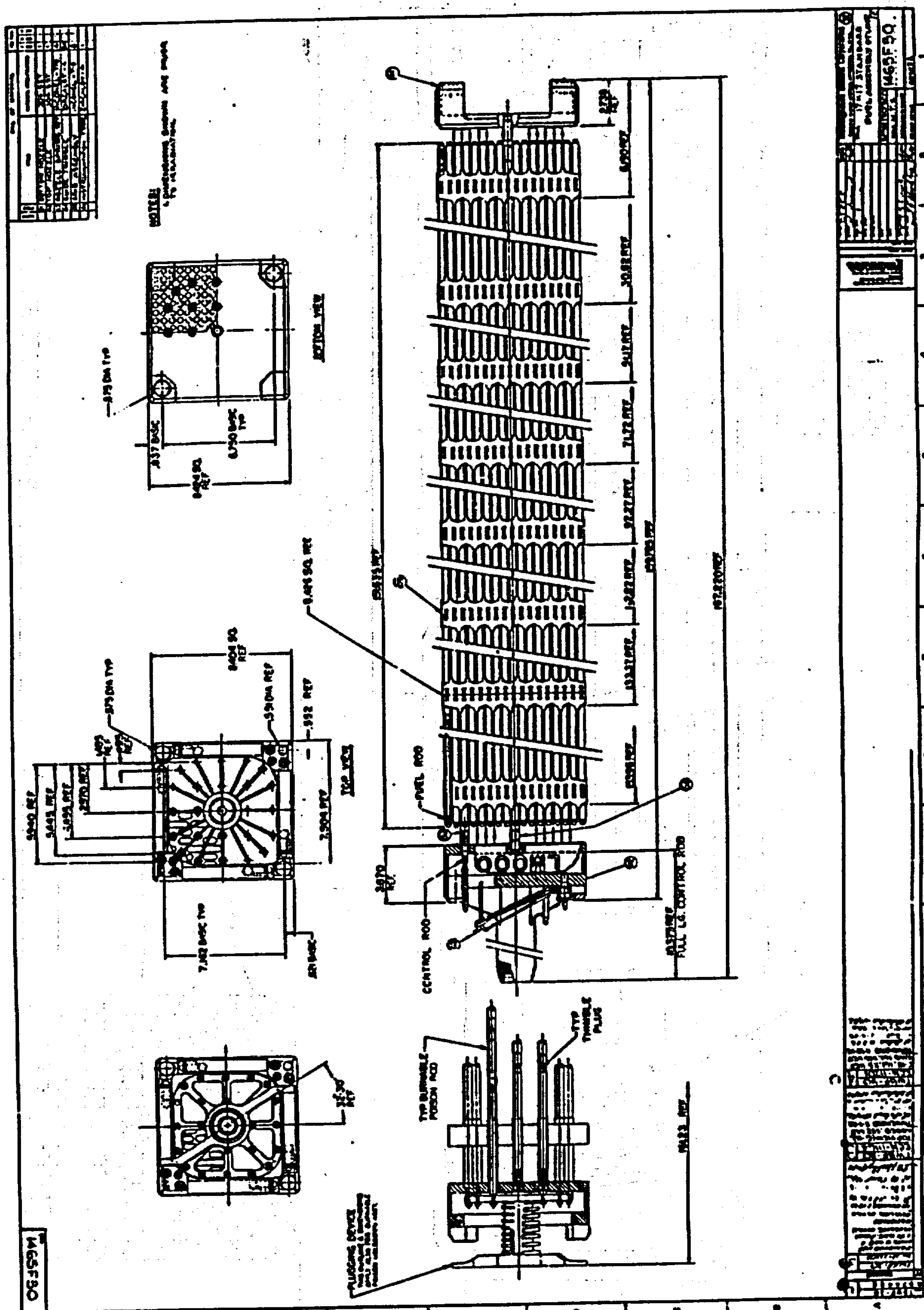
B-2. Fuel Assembly Drawing (Calvert Cliffs Units 1 and 2, Millstone Unit 2, Maine Yankee, St. Lucie Unit 1)



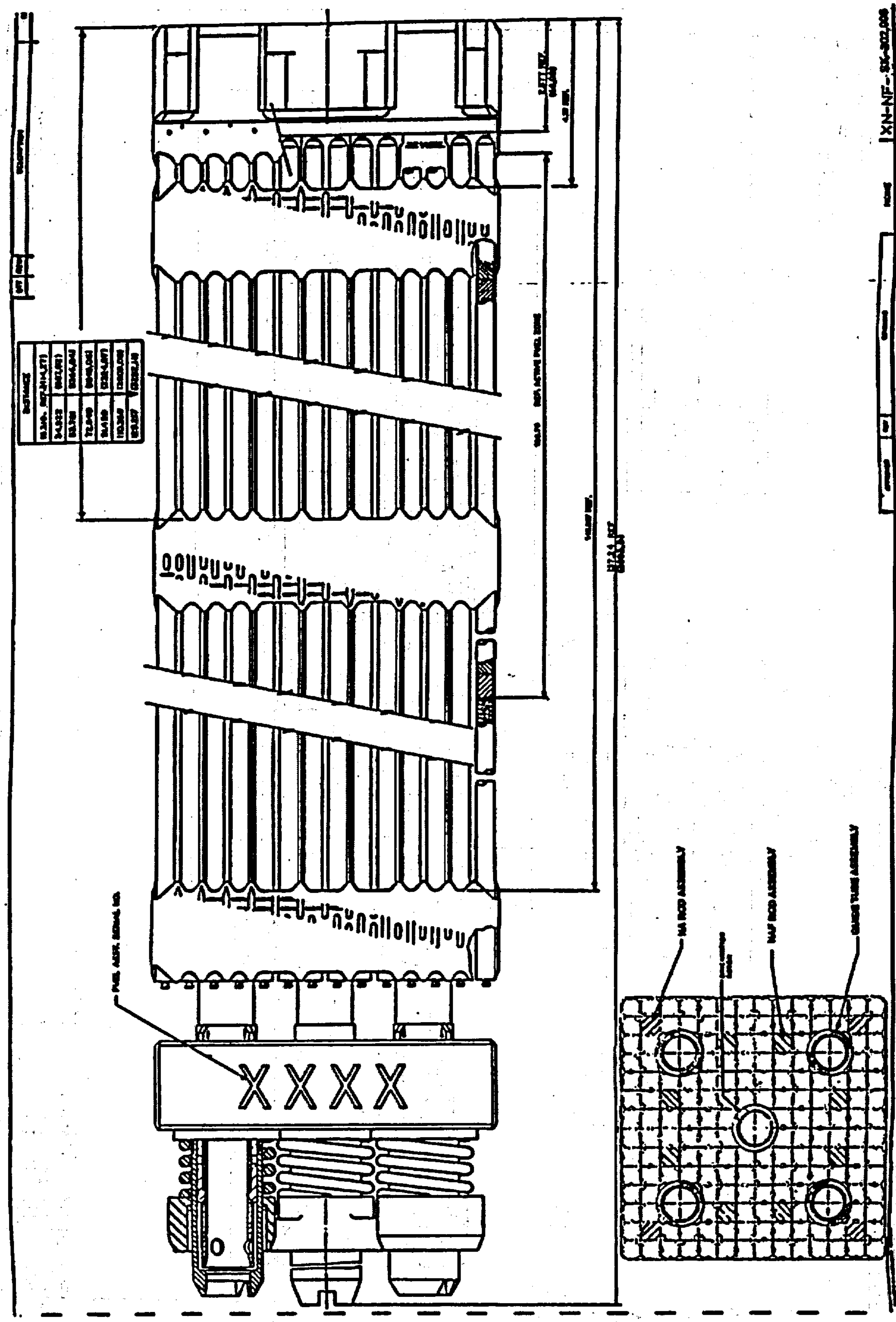
B-3. Fuel Assembly Drawing [Palo Verde Units 1, 2 and 3 (System 80)]



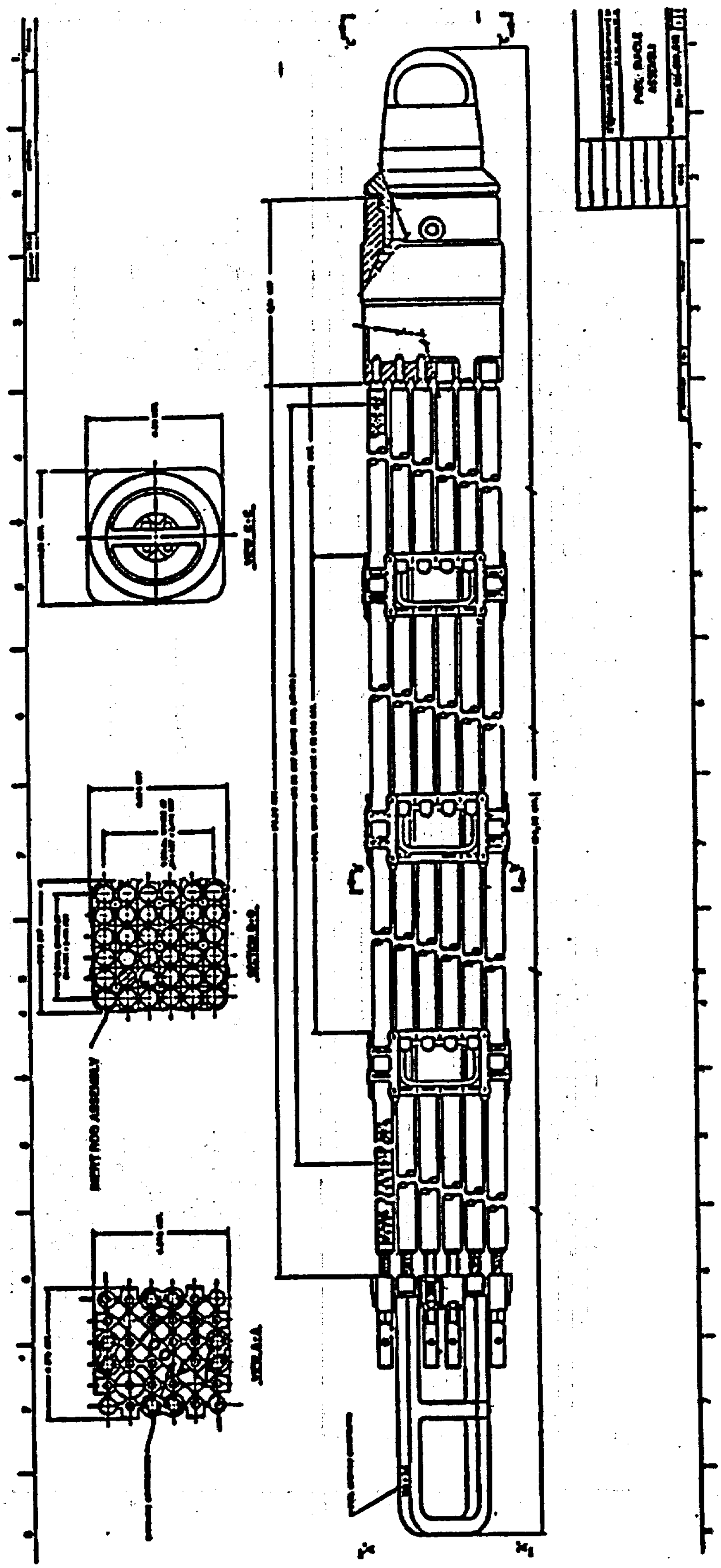
B-4. Fuel Assembly Outline 17 x 17 [(Conceptual), (South Texas Project Units 1 and 2)]



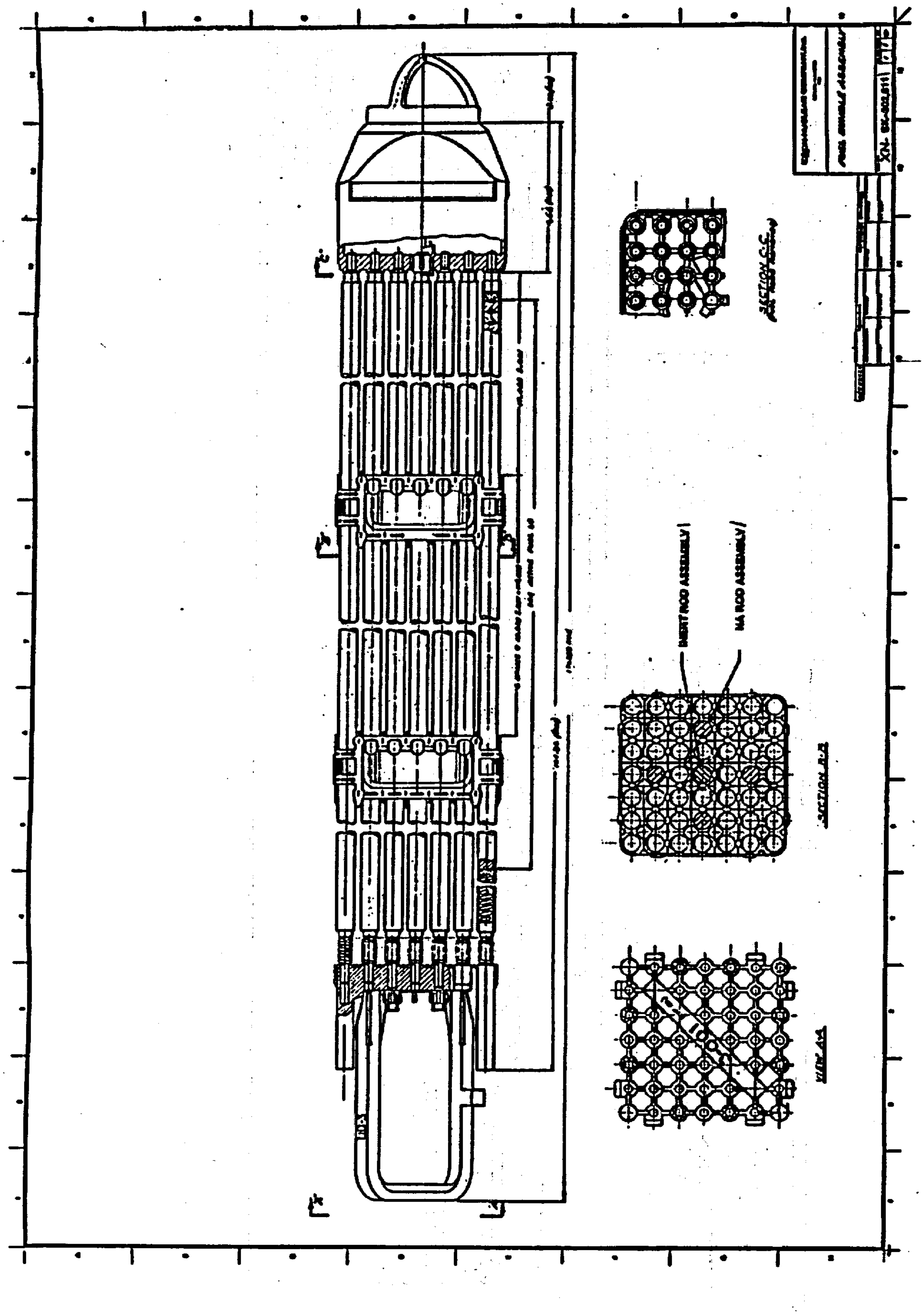
B-5. Fuel Assembly Outline 17 x 17 (Standard)



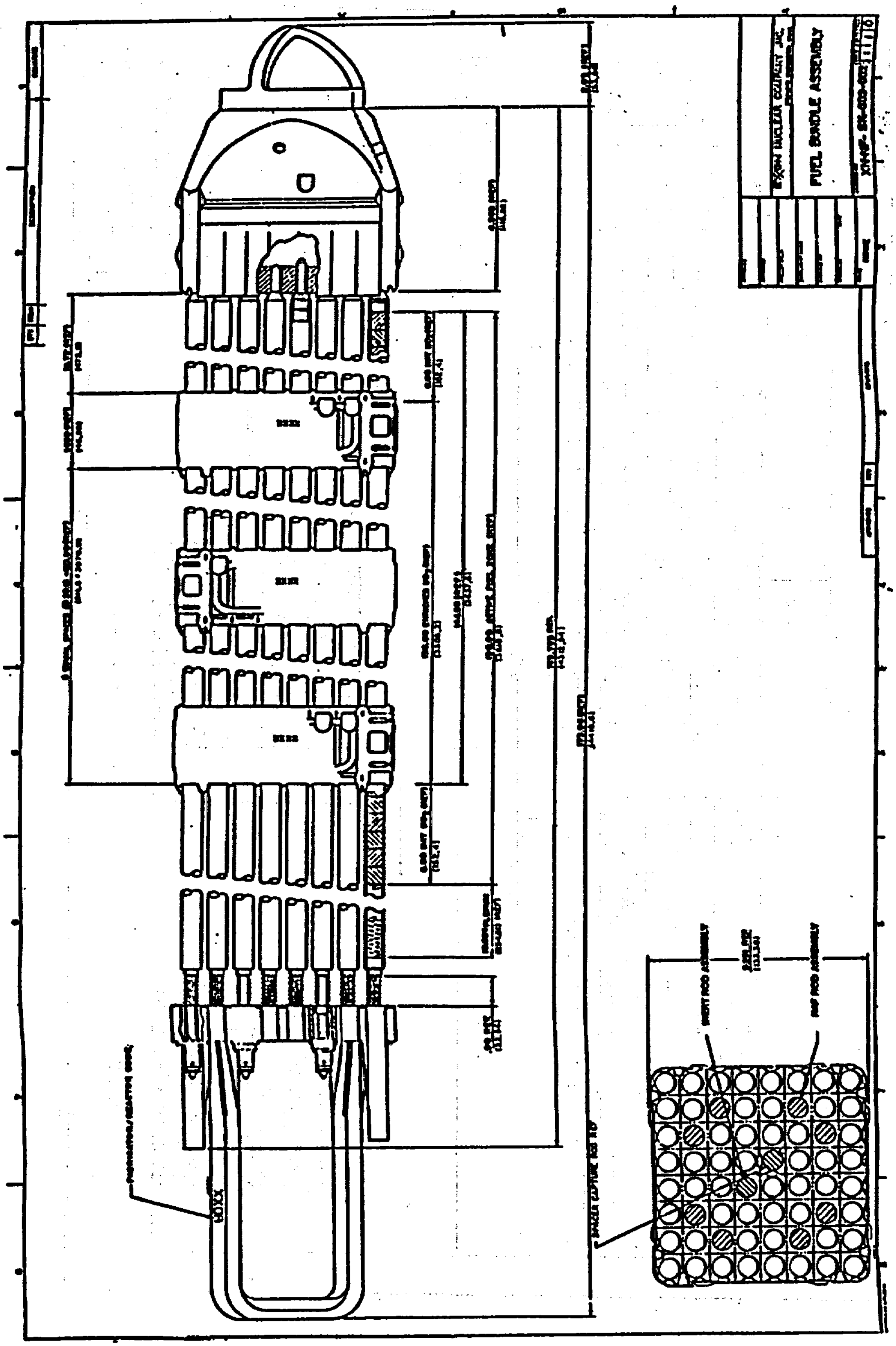
B-6. Fuel Bundle Assembly (XN-NF-SK-302,006)



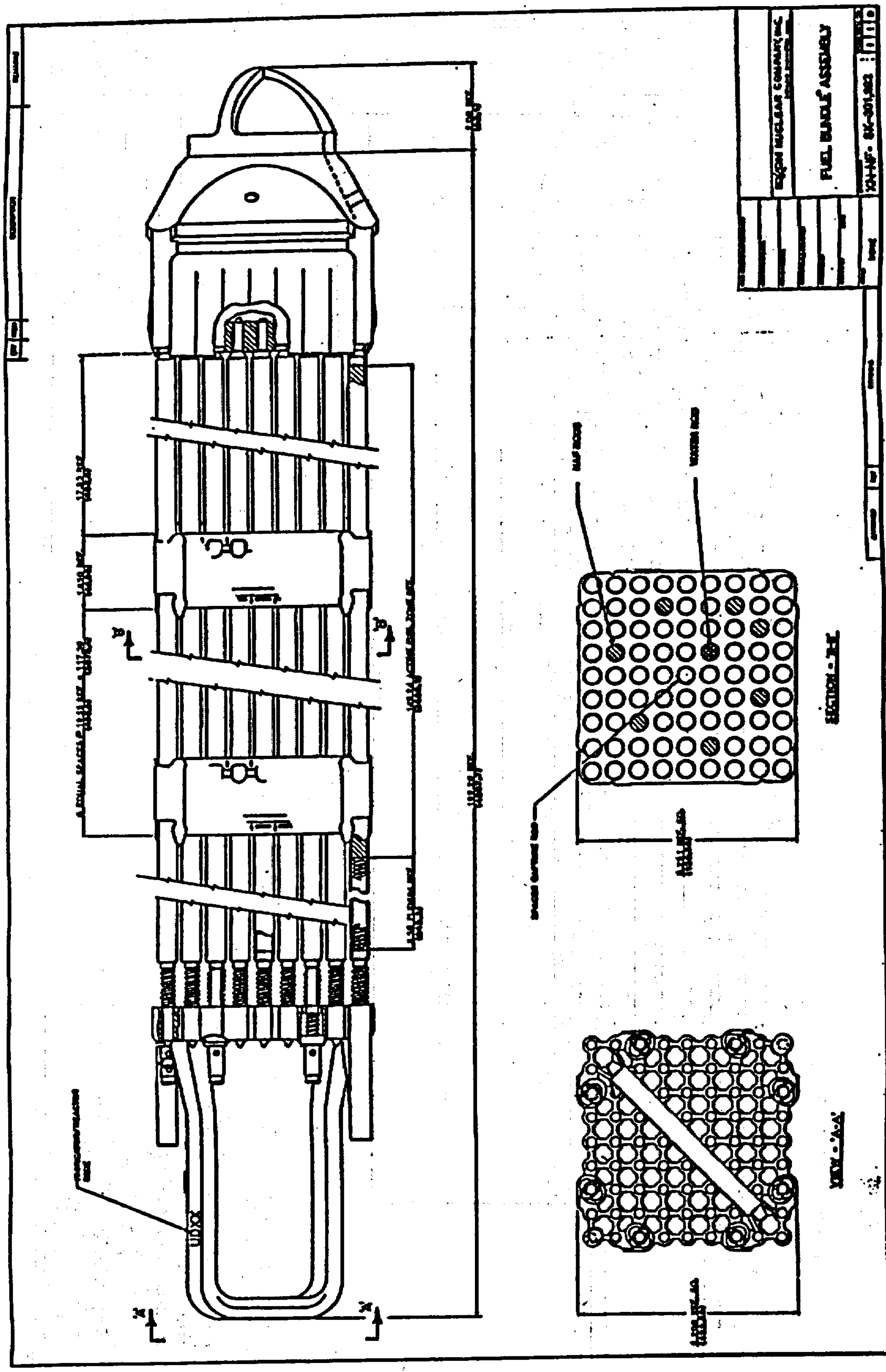
B-7. Fuel Bundle Assembly (XN-SK-302,012)



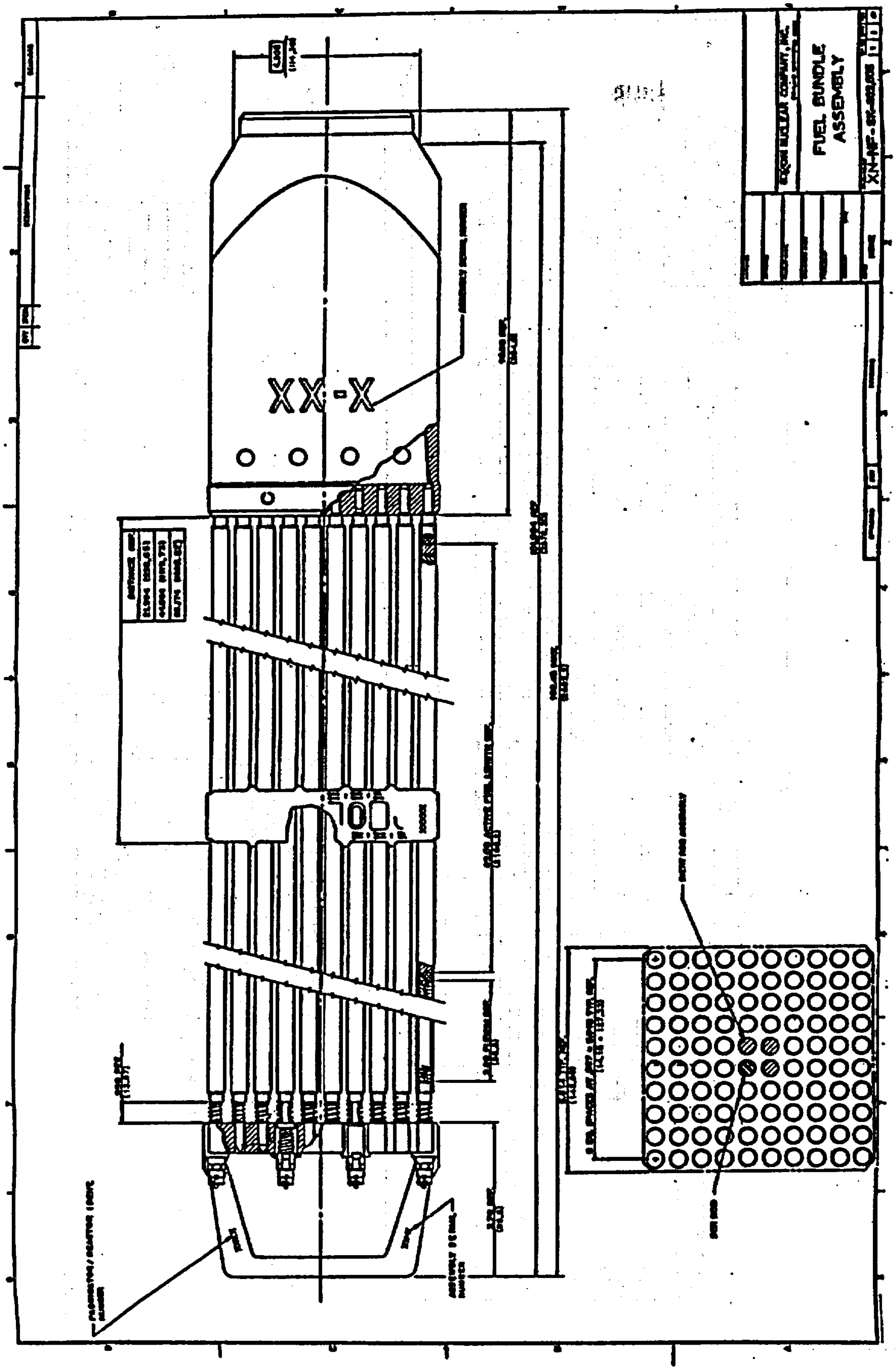
B-8. Fuel Bundle Assembly (XN-SK-302,011)



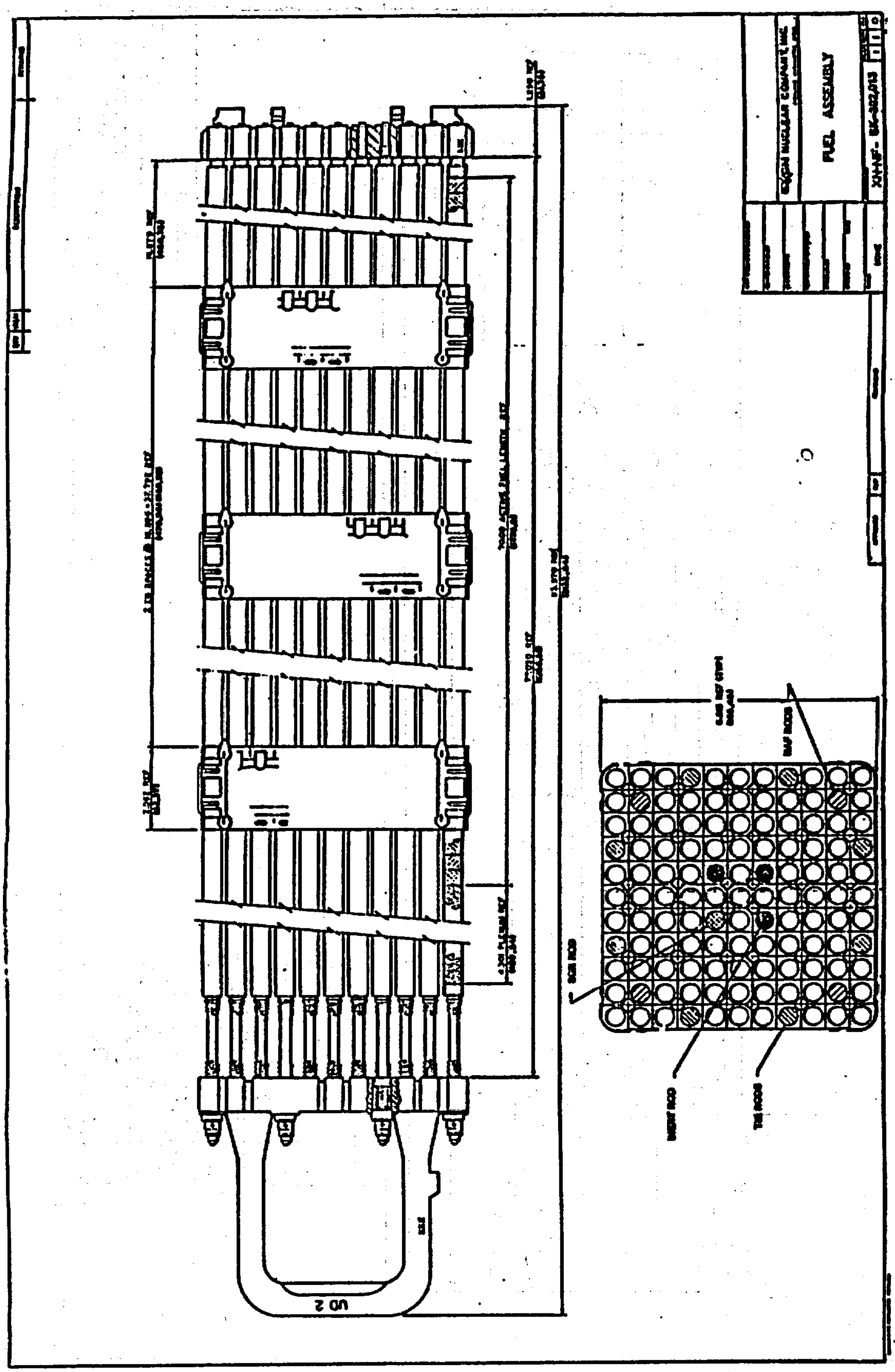
B-9. Fuel Bundle Assembly (XN-NF-SK-302,002)



B-10. Fuel Bundle Assembly (XN-NF-SK-301,992)



B-11. Fuel Bundle Assembly (XN-NF-SK-302,005)



B-12. Fuel Assembly (XN-NF-SK-302,013)

APPENDIX C

CANISTERED FUEL PHYSICAL CHARACTERISTICS

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CANISTERED FUEL PHYSICAL CHARACTERISTICS

C.1 CANISTER CHARACTERISTICS NOT EXPECTED TO VARY OVER TIME

C.1.1 NONDISPOSABLE CANISTERS (SINGLE AND DUAL-PURPOSE CANISTERS)

Information to be developed for subsequent revision.

C.1.2 PROBLEMATIC SNF IN DISPOSABLE CANISTERS

Categories of CSNF projected to be placed into disposable canisters (defined as canisters that can be placed into disposal containers without being repackaged) include:

- Mechanically and Cladding-Penetration Damaged SNF - SNF classified by waste Purchasers as failed because of (1) mechanical damage that limits the ability to vertically lift the assembly, or to fit within the dimensional envelope of standard fuel, and/or (2) "known or suspected cladding defects greater than a hairline crack or pinhole leak" as defined in the NRC Staff Guidance on Damaged Fuel (NRC 1998).
- Consolidated/Reconstituted Assemblies - SNF assemblies that were disassembled and, if reassembled, were done so in a form that is dimensionally different from the original.
- Fuel Rods, Pieces, and Debris - Variable-sized pieces of fuel (ranging from a single pellet to a full rod) and debris combining fuel and nonfuel materials.
- Nonfuel Components - Includes in-core assembly components physically separated from the assemblies and shipped separately.

Assumptions used in the estimates provided below include:

- Existing canistered waste (excluding multi-element canisters) can be transported in accordance with 10 CFR 71 and will not be repackaged by the Purchaser.
- All canistered failed assemblies and fuel debris are in either unsealed screen-end canisters or sealed solid-end canisters, both comparable in size to that of uncanistered assemblies (i.e., they fit into racks in reactor pools). No sealed canisters will be used for failed assemblies.
- All canisters can be handled like uncanistered SNF at that reactor site (similar crane hook interface).
- Existing/projected non-integral nonfuel components will be delivered to the CRWMS (not sent to a low-level waste disposal facility, as is currently done by certain utilities).
- All but a small fraction of nonfuel components will remain integral to the assembly (excluding non-integral items physically removed from the assembly to meet the assembly dimensional or weight requirements for transport).

C.1.2.1 Existing Quantities of Canistered SNF Assumed to Be Disposable

Materials currently canistered for containment or handling reasons will be delivered "as is" unless 10 CFR 71 precludes transport. Given the assumption that all existing canistered wastes will meet 10 CFR 71 and will not be repackaged prior to transport, this means currently canistered materials are treated as disposable in this report. This disposability assumption does not apply to multi-element canisters used at utility sites for long-term storage.

Specific canister-handling data are currently unavailable in CRWMS databases. This includes the physical construction of the canister (e.g., are the ends screened?), the degree to which these canisters can be loaded in a disposal container without repackaging (e.g., are free liquids removed?), and the ease with which the canister can be handled. (Are handling fixtures permanently attached, how easily are they attached if not permanently part of the assembly, and what impact does the addition of the handling fixture have on the overall canister dimensions?) Existing information in CRWMS databases include the number of canisters of waste, the canister physical dimensions, and a qualitative assessment of canister contents relative to the canistered SNF categories described above. This section covers existing canisters of wastes. Existing (but currently uncanistered) wastes to be canistered before transport, or wastes to be generated and canistered in the future, are addressed in Section C.1.2.2.

Data reported by Purchasers through December 31, 1994, list 265 canisters of material in commercial inventories (DOE 1996). These canisters are subdivided as follows:

- 145 canisters of fuel only (intact assemblies, intact rods, rod pieces, pellets, fuel debris)
- 101 canisters of nonfuel components only
- 19 canisters of a mixture of fuel and nonfuel materials.

Canisters containing only fuel or a mixture of fuel and nonfuel components are equally distributed between those containing an assembly and those with only "loose" fuel (ranging from intact rods to debris). Precise counts are 42 canisters with intact assemblies, 87 canisters with rods and pieces (up to 31 may contain an assembly plus rods/pieces), 31 canisters with consolidated assemblies, and 4 canisters with "unknown" contents (DOE 1996, Table 16).

Dimensionally, these canisters are diverse but fit within assembly dimensional envelopes. Most range in cross-section from 5 x 5 inches to 9 x 9 inches, and in length from 138 inches to 189 inches. There are 13 canisters with no dimensions reported (all are assumed to fall within the 5 to 9 inch and 138 to 189 inch envelope). Various sizes are widely distributed among Purchaser sites. Table C-1 lists canisters with reported dimensions outside these ranges.

Table C-1. Existing Canisters with "Non-Standard" Dimensions [a]

Reactor Name	Number of Canisters	Canister Contents	Canister Length (inches)	Cross-Section (inches)
Palo Verde 1 & 2	2	Fuel Rods	180	1 x 1
Arkansas Nuclear 2	1	Fuel Rods/Pieces	176	16 x 16
Calvert Cliffs 1 & 2	16	Nonfuel Debris	148	2 x 2
Byron 1 & 2	4	Assemblies	170	19 x19
Big Rock Point	3	Partial Fuel Rods	60	12 x 12
Big Rock Point	2	Fuel Pieces	18	12 x 12
Big Rock Point	2	Fuel Rods	84	6 x 0[b]
Fermi 2	4	Nonfuel Components	180	36 x 0[b]
McGuire 1	1	Single Fuel Rod	170	10 x 10
Beaver Valley	2	Fuel Rods	170	14 x 14
Oconee 1 & 2	6	Fuel Rods	170	10 x 10
Millstone 2	2	Single Fuel Rod	171	3 x 3
Prairie Island 1 & 2	2	Nonfuel Components	159	5 x 5
Humboldt Bay	1	Assembly	84	12 x 12
Point Beach 1 & 2	1	Single Fuel Rod	180	10 x 10
Total Fuel	27			
Total Nonfuel	22			

NOTES: [a] DOE 1995b.
[b] Devotes a cylindrical container, with diameter = the non-zero value.

It should be assumed that all existing canisters that meet the following criteria will be transported "as is":

- Meet the dimensional requirements for baskets in existing or projected transport casks
- Have screened ends (assumed to be all canisters in pool storage)
- Have intact lifting fixtures that allow the canister to be safely lifted vertically by the same equipment used to lift uncanistered intact assemblies.

C.1.2.2 PROJECTED QUANTITIES OF CANISTERED SNF

Projected quantities of canistered SNF include existing materials not yet canistered (e.g., to meet 10 CFR 71 requirements for transportation) and SNF projected to be generated and canistered in the future. For existing materials, projections depend on which "failed" assemblies need to be canistered. These projections easily vary by two orders of magnitude depending on how failure modes are classified. A recent proposed classification (EPRI 1997) groups failure modes into those involving assembly mechanical damage (failure of handling fixtures or distortions in assembly dimensions), and cladding penetration and damage. Both failure modes are further subdivided into minor and major. Assemblies with "major" mechanical or cladding failure are rare and must be canistered to be handled at the Purchaser site. They also must be canistered for transport, as the NRC routinely includes in transport cask certificates of compliance the provision that failed assemblies (other than those with pinhole leaks or minor cracks) be canistered prior to transport.

Some fraction of failures are borderline, and it is recognized that most failed assemblies may be stored for a number of years. It is possible that a current inconsequential failure may become more problematic over time. Data on the long-term degradation of "failed" assemblies aren't definitive, so estimates of the number of these assemblies that will be "major" failures at the time of transport are uncertain. There is also the issue of assembly damage during cask loading. It is assumed, for the purposes of this study, that 15 percent of the assemblies currently identified as "failed," but whose failure is currently considered minor, will be canistered prior to transport. EPRI, 1997, estimates 10-20 percent of assemblies designated as failed to be "intermediate" between pinhole/hairline-crack failure and visible failure. This estimate covers any assemblies damaged during handling activities at the Purchaser site.

C.1.2.2.1 Existing Uncanistered "Failed" Fuel Requiring Canistering

Estimates of existing failed assemblies are 4,864 to 9,728 assemblies (EPRI 1997). Of these, roughly 50-100 are severe failures (EPRI 1997) and are assumed to be part of the existing canistered-waste inventory. Subtracting the severe failures and applying the 15 percent intermediate-failure assumption yields 700-1,400 assemblies. It is assumed, for the purposes of this study, that 1,100 existing assemblies will have sufficient mechanical or cladding damage to justify canistering. It should be noted, however, that this number represents a range of 500 (roughly 10 percent of 4,864) to 1,950 (roughly 20 percent of 9,728), and that any conceivable fraction of these "intermediate-failure" assemblies may be safely handled and transported without being canistered. It is assumed that these assemblies will be placed into unsealed single-element canisters.

C.1.2.2.2 Projected Canistered "Failed" Fuel

Current and projected trends in Purchaser handling of "failed" assemblies are based only on the more recent operational data because of improved operations over time. For example, failed assemblies discharged prior to 1986 averaged 2.2 failed rods per assembly (EPRI 1997), which suggests that the vast majority of the fuel cladding in the assembly remains intact. More recently, Purchasers are replacing failed rods, reinserting the "repaired" assembly, and collecting the failed fuel into canisters or reconstituting them into assemblies with exclusively failed rods. Whereas most historical failed assemblies are "minor" failures, the majority of projected failed assemblies will have more extensive problems.

Rough projections of the "to be generated" failed CSNF are based on the following assumptions:

- There will be no significant improvement in assembly failure rates in the future (use of current failure rates to generate projections is conservative but reasonable).

- The split between canisters containing intact assemblies and loose rods or fuel debris will continue to reflect historical trends (a roughly 50-50 split).

- Canisters containing individual rods and fuel debris will contain the equivalent of 20 percent of the fuel in a full assembly.

- An additional 46,900 MTU will be generated between 1998 and 2035 (36,900 MTU exists now, for an estimated total of 83,800 MTU in 2035). [*Calculation Method for the Projection of Future SNF Discharges* (Draft). A000000000-01717-0200-0052 REV 00, Vienna Virginia: CRWMS M&O. ACC: MOV.19990625.0001.]

- Each of the 104 currently operating reactors will have some failed rods, and reactors that share a pool can combine failed rods into a single canister or reconstituted assembly.

- Failed fuel will be shipped to the repository from each Purchaser storage site four times over the waste-emplacement life of the repository (roughly once every 10 years).

Reference EPRI 1997 states that the current fuel rod failure rate is about 0.01 percent (versus 0.02 - 0.07 percent in the first 20 years of commercial nuclear power). The projected 46,900 MTU of SNF to be generated translates into 163,700 assemblies [*Calculation Method for the Projection of Future SNF Discharges* (Draft). A000000000-01717-0200-0052 REV 00. Vienna, Virginia: CRWMS M&O. ACC: MOV.19990625.0001]. If all failed rods are removed from the original assemblies and reconstituted as failed assemblies without regard to time or geographic location, this would yield only 16-17 failed-assembly equivalents (163,700 assemblies x 0.01 percent). Note that this represents a theoretical minimum only, as there are 104 operating reactors, and failures will be distributed over each reactor's remaining operating life. As a point of reference, distributing the failed-assembly equivalents evenly among the 72 existing storage sites yields 0.23 assembly-equivalents per site (16.37 assemblies divided by 72 sites). The following logic results in a more realistic estimate.

- There will be a minimum of four reconstituted assemblies (using only failed rods) per storage site. This results in 288 reconstituted assemblies, to address "minor" cladding penetration damage (reasonable given that the theoretical minimum site average is only 0.23 canister equivalents over 40 years).

- There will be an average of two canisters of fuel debris per site per each 10 years of repository emplacement operations (8 per site times 72 sites). This is based on the assumption that some of the damaged rods will simply be placed into a canister, rather than reconstituted into the "failed-fuel" assembly addressed in the previous bullet.

- There will be an average of 2.3 canisters per site for assemblies with "major" mechanical damage (corresponds to roughly one-tenth of 1 percent of the 163,700 assemblies to be generated) or for assemblies identified as failed and simply discharged.

This yields a total projection of approximately 1,000 unsealed canisters of failed fuel to be delivered to the CRWMS (14.3 canisters per site times 72 sites). It can be assumed that all of these canisters will be screened-end containment vessels essentially identical to those currently in use and dimensionally compatible with an uncanistered assembly. Given that cask owners have the option of using sealed canisters (none currently reside in commercial utility pools), some small fraction of the projected 1,000 canisters may be sealed (solid end) canisters that need handling through the canister transfer system.

It should be noted that the total projection of approximately 1,000 canisters of failed fuel is consistent with the historical data. Assuming 46,900 MTU of SNF will be generated and 36,900 MTU has been generated to date, the ratio of generated to projected is 46.9 to 36.9 or 1.27 to 1. There are currently 164 existing canisters containing fuel, a maximum of 300 canister equivalents of fuel currently reported as "in baskets," and 1,100 assemblies designated as "failed" (see Section C.1.2.2.1). Applying the 1.27:1 ratio yields roughly 2,000 projected canisters. Recognizing that there was a much higher assembly failure prior to 1981 and more at-reactor research/testing that generated wastes needing canistering, a ratio of roughly 2:1 between existing and projected inventories appears reasonable.

C.1.2.2.3 Projected Canisters of Other Existing Uncanistered Fuel

Data reported by Purchasers through December 31, 1994, indicate that there are 300 "baskets" of material in commercial inventories (DOE 1996). These are primarily open baskets in reactor pools, although some are referred to as "sealed baskets." Data on these baskets are incomplete, and it cannot be determined what quantities of materials are in each basket, the distribution between fuel and nonfuel components in each basket, or which of these might be double-counted canisters. Most likely, the contents of each basket will be placed into its own canister, with all canisters being the screened-end containment vessels essentially identical to those currently in use and dimensionally compatible with an uncanistered assembly. This translates to 300 canisters of currently "basketed" waste that will be delivered to the CRWMS. The alternative is that the SNF will be shipped in these "baskets" (this assumes that they are shippable per 10 CFR 71, which is unlikely). This indicates that the CRWMS may receive sealed baskets ranging in size from 12x12x36 (inches) to 1x1x180 (inches).

There are also a small number of "disassembled and reconfigured" assemblies (product of consolidation trials) that are currently stored uncanistered. Some are reconfigured into a larger assembly (two such assemblies exist at Maine Yankee) that still meet the dimensional definition of standard but don't match any commercial fuel design. These assemblies are not specifically identified by the Purchasers in the RW-859 data sets (DOE 1996); therefore, there is no estimate on their number or condition. It is assumed that any existing uncanistered "reconfigured" assemblies can be safely handled "as is" and that any new "reconfigured" assemblies will be transported in canisters.

C.1.2.2.4 Projected Canisters of Nonfuel Components (Existing and Projected Nonfuel Components)

Three variables significantly influence the estimated number of nonfuel component canisters transported, with nonfuel components defined in (DOE 1996) as:

- Components used to initiate, control, and monitor the chain reaction in the core (neutron sources, control elements, burnable absorbers, in-core instrumentation, etc.).

- The nonfuel portion of a fuel assembly (often called guide tubes, water rods, grids, nozzles, etc.).

- Miscellaneous hardware used in the reactor core that is not a part of fuel assemblies (dummy assemblies, coupon trees, thimble plugs, etc.).

The first variable involves the degree to which these components remain integral to the assembly. The vast majority can remain integral. The EIA (DOE 1996) reports that approximately 91 percent of the nonfuel components are currently stored integral to the assemblies. There is usually a slight increase in assembly dimensions (size and weight) due to integral nonfuel components (e.g., adds 100-200 lbs. to the assembly weight); however, these components are already addressed in Section B.1.1 of this report. It is assumed that the maximum number of nonfuel components possible will remain integral to the assembly. It should be noted, however, that the estimated number of nonfuel canisters increases by a factor of 2 to 10 if nonfuel components are removed from the assembly and shipped separately (CRWMS M&O 1992).

The second important variable is the degree to which these materials are managed as low-level waste and never delivered to the CRWMS (e.g., Purchaser chooses to create additional pool storage by shipping these materials to a low-level waste disposal facility). It is conservatively assumed that all existing nonfuel components, except those at sites with exhausted pool storage prior to January 30, 1998, will be delivered to the CRWMS. It is assumed that these sites chose to make additional storage space available by shipping these materials to low-level disposal facilities rather than purchase additional dry storage capacity.

The third significant variable involves the method for canistering the non-integral nonfuel components. Different volume-reduction methods can significantly affect the number of components per canister and hence the total number of canisters. Estimates of the number of canisters containing nonfuel components easily vary by a factor of five or more under compacted versus uncompacted scenarios (CRWMS M&O 1992).

Purchasers reported 65,190 pieces of nonfuel components from 104,742 discharged assemblies through December 31, 1994 (DOE 1996). Of these components, approximately 6,500 pieces are reported as non-integral to the assemblies. Using the same 1.27:1 ratio of projected to actual discharges, this translates to a projected additional 8,300 non-integral nonfuel components to be generated by 2040. It must be recognized, however, that:

1. Some fraction of the existing materials are already canistered in the 101 reported canisters of nonfuel components and the 19 reported canisters of mixed fuel and nonfuel components.

2. Another fraction of these materials currently reside in the 300 baskets reported in reactor pool inventories.

3. Once the repository becomes operational, there will be little incentive to follow historical practices of shipping some fraction of this material to low-level waste disposal facilities. Therefore, total quantities of materials delivered to the CRWMS likely will increase.

The study *System Aspects of Non-Fuel Bearing Hardware within the CRWMS* (CRWMS M&O 1992) estimated the quantities of nonfuel components for 63,000 MTU of fuel under three scenarios (maximum integral nonfuel components, compacted but all non-integral, and uncompacted non-integral). The maximum integral case assumed that only the control-rod assemblies from CE System 80 PWR assemblies (Palo Verde), BWR control assemblies (cruciforms and bases), BWR neutron-source assemblies, and BWR instrumentation assemblies are non-integral. For the purposes of this study, the same assumption will be used, with the exception that it also is assumed that 1 percent of BWR fuel channels [matches historical percentage if Nine Mile Point is excluded (DOE 1996)] and 4.5 percent of all other nonfuel components are also non-integral [matches historical percentage of non-integral PWR components (DOE 1996)].

Extrapolating data from Table 1 in *System Aspects of Non-Fuel Bearing Hardware within the CRWMS* (CRWMS M&O 1992) yields estimated total quantities of canistered, non-integral nonfuel components, assuming existing plus projected inventories of 125,000 PWR assemblies and 167,000 BWR assemblies [*Calculation Method for the Projection of Future SNF Discharges* (Draft). A000000000-01717-0200-0052 REV 00. Vienna, Virginia: CRWMS M&O. ACC: MOV.19990625.0001]. These estimates are summarized in Table C-2.

Table C-2. Estimated Total Canisters of Nonfuel Components

Component Description	Projected Components	Dimension of Can (Inches)	Components per Can	Total Cans
PWR Control Assy (CE Sys. 80) - Rod Sets	720 [c]	9x9 x 160	17	43
PWR Control Ass'y (CE Sys. 80) - Spiders	720 [c]	9x9 x 160	25	29
Other Misc. PWR Components [a]	5,625 [d]	9x9 x 160	12 [d]	470
BWR Control Assy - Cruciforms	5,400 [e]	9x9 x 160 [e]	2 [e]	~2,700 [e]
BWR Control Assy - Bases	5,400 [e]	10.6 (dia) x 177	8	680
BWR Neutron Source Ass'y	295 [f]	6x6 x 160	7	42
BWR Instrumentation Ass'y	2,270 [g]	6x6 x 160	44	52
Other Misc. BWR Components [b]	1,670 [h]	6x6 x 168	7	240
TOTAL	21,700			~4,300 [i]

NOTES: [a] Includes control-rod assemblies other than those for CE System 80, burnable poison assemblies (rods and spiders), neutron source assemblies (rods and spiders), in-core instruments, and thimble plug assemblies (rods and spiders).
[b] Includes fuel channels.
[c] Assumes 1 control assembly for each 10 Palo Verde assemblies discharged (approximately 7,200 projected through 2040).

Table C-2. Estimated Total Canisters of Nonfuel Components (Continued)

[d] Represents 4.5 percent of assumed total 125,000 discharged PWR assemblies through 2035, and a canister loading of 12 pieces per canister.
[e] Assumes ratio of 1 control assembly for each 100 discharged BWR assemblies prior to 1998 (total of 700 control assemblies), and 5 control assemblies for each 100 discharged BWR assemblies thereafter (total of 4,700); also assumes that cruciforms will be cut into two L-shaped pieces and stacked so that four cut halves fit into a 9 in. by 9 in. canister (placement into larger cans will reduce total number of canisters).
[f] Assumes historical ratio of 1.77 neutron sources per 1,000 discharged BWR assemblies (DOE 1996) for projected 167,000 total BWR assemblies.
[g] Assumes historical ratio of 1.36 instrumentation assemblies per 100 discharged BWR assemblies (DOE 1996) for projected 167,000 total BWR assemblies.
[h] Uses 1 percent assumption and projected total BWR discharge of 167,000 assemblies.
[i] Should be reduced by 100 to account for existing canisters of nonfuel components.

Loaded canisters are projected to weigh from roughly 300 lbs. for BWR neutron source assemblies to approximately 2,500 lbs. for PWR control-rod-assembly rod sets (CRWMS M&O 1992).

C.2 PARAMETERS PROJECTED TO VARY OVER TIME

There are no data indicating whether single-element-sized canisters of SNF will be given priority shipment (i.e., remove the more problematic SNF from a site first) or left for last (not shipped until canisters are filled as much as possible before shipment). Given that these canisters are dimensionally interchangeable with bare assemblies, MGR annual receipt rates or variations in these rates should have no impact on facility design unless there is a different protocol or greater difficulty in drying these canisters prior to loading into disposal containers.

There is no design basis receipt rate assumed at this time for multi-element disposable canisters.

C.3 ADDITIONAL DESIGN CONSIDERATIONS

As with off-normal physical conditions for assemblies, there is a remote possibility that single-element-size canisters loaded undamaged into transport casks will either be jammed in the cask basket or otherwise damaged during transport. The conservative assumption that should be used in design is that there will be one cask shipment per year with at least one damaged canister that was undamaged prior to transport. An additional conservative assumption that should be used in design is that biennially one cask shipment will have a problem that requires the cask be taken off-line and remediated on a case-by-case basis.

There is some question regarding existing loaded storage canisters being granted a 10 CFR 71 exemption to enable them to be transported without being repackaged. These canisters are dimensionally similar to canisters licensed as DPCs; therefore, there should be no special consideration required relative to Surface Facility design.

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APPENDIX D

TRANSPORT CASK PHYSICAL CHARACTERISTICS

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TRANSPORT CASK PHYSICAL CHARACTERISTICS

D.1 CASK CHARACTERISTICS NOT EXPECTED TO VARY OVER TIME

Four types of cask/carrier systems can be expected: a small truck cask that contains a single PWR assembly or two BWR assemblies, a larger truck cask that contains up to four PWR assemblies or nine BWR assemblies, rail casks on a rail car, or rail casks on a heavy-haul truck. Both the small and larger truck casks are transported via legal weight truck.

D.1.1 DIMENSIONS AND WEIGHTS

D.1.1.1 Carriers

See Table 7 in Section 3.3 for a description of cask carrier systems bounding physical characteristics.

D.1.1.2 Casks

Table D-1. Cask Physical Characteristics by Cask Type[a]

Cask Type	Description	Length (in.)[b]		Diameter (in.)[b]		
		I: Yes	I: No	T: Yes I: Yes	T: Yes I: No	T: No I: No
			TRUCK			
1	NAC LWT[c]	231.8	199.8	60.3	44.2	44.2
2	GA-4[d]	233.8	187.8	90.0	48.0	39.8
3	GA-9[e]	244.5	198.0	90.0	47.7	39.8
			RAIL			
4	Generic Small[p]	TBD	210.0	TBD	TBD	TBD
5	Generic Medium/WESFLEX[f, g]	295.8	205.3	126.5	102.8	87.8
6	Generic Large/HISTAR-100[h]	305.9	203.1	128.0	96.0	96.0
7	Small HH[p]	TBD	210.0	TBD	TBD	TBD
8	Medium HH[p]	TBD	210.0	TBD	TBD	78.7
9	South Texas[o]	340 (TBV)	233.2	TBD	TBD	TBD
10	South Texas HH[o]	340 (TBV)	233.2	TBD	TBD	TBD
11	PWR West Valley[i]	234	~180	131.0	TBD	~90.5
12	BWR West Valley[j]	244.5	190.5	131.0	TBD	83.25
13	TranStor[k]	326.7	210.0	140.0	92.3	92.3
14	MP-187[l]	307.5	201.5	126.8	92.5	92.5
15	NAC UMS[m]	271.3	209.3	124.0	92.9	94.3

Table D-1. Cask Physical Characteristics by Cask Type[a] (Continued)

Cask Type	Description	Length (in.)[b]		Diameter (in.)[b]		
		I: Yes	I: No	T: Yes I: Yes	T: Yes I: No	T: No I: No
RAIL						
16	Yankee Rowe[n]	257.0	193.0	124.0	99.0	99.0
17	Big Rock Pt[f, g]	295.8	205.3	126.5	102.8	87.8
18	HLW Short	TBD	TBD	TBD	TBD	TBD
19	HLW Long	TBD	TBD	TBD	TBD	TBD

NOTES: [a] Cask types reflect those in Table A-1 of Appendix A.
[b] Abbreviations under sub-columns are as follows:
T: Yes = Cask with trunnions
T: No = Cask without trunnions
I: Yes = Cask with impact limiters
I: No = Cask without impact limiters
All Dimensions rounded to nearest tenth of an inch.

SOURCES: [c] Nuclear Assurance Corp., Safety Analysis Report for the NAC Legal Weight Truck Cask , Revision 13, NAC T-88004, Docket No. 71-9225, March 1995 (NAC 1995).
[d] General Atomics (GA), GA-4 Legal Weight Truck Spent Fuel Shipping Cask Safety Analysis Report for Packaging (SARP), SAR-71-9226, January 31, 1997 (GA 1997).
[e] General Atomics (GA), GA-9 Legal Weight Truck Spent Fuel Shipping Cask Safety Analysis Report for Packaging (SARP), SAR-71-9221, July, 1994 (GA 1994).
[f] Westinghouse Electric Co., Wesflex™ Storage System Safety Analysis Report, Revision 0, WSNF-200, Docket No. 72-1026, February 1998 (WEC 1998).
[g] Westinghouse Government and Environmental Services Corp., Safety Analysis Report, Large Transportation Cask Subsystem, Multi-Purpose Canister Project, MPC-CD-02-014, Revision 2, October, 1996 (WGESC 1996).
[h] Holtec International, Safety Analysis Report for the Holtec International Storage, Transport and Repository (HI-STAR) 100 Cask System, Revision 8, Holtec Report No. HI-951251, Docket No. 71-9261, February 20, 1999 (Holtec 1999).
[i] U.S. Nuclear Regulatory Commission Certificate of Compliance for Radioactive Materials Packages, Model No. TN-REG, Certificate 9206, Revision 4, March 26, 1996 (NRC 1996, pp. 356-359).
[j] U.S. Nuclear Regulatory Commission Certificate of Compliance for Radioactive Materials Packages, Model No. TN-BRP, Certificate 9202, Revision 5, March 26, 1996 (NRC 1996, pp. 344-347).
[k] BNFL Fuel Solutions (BFS) Safety Analysis Report for the TranStor™ Shipping Cask System, SAR, Revision A, February 1999 (BFS 1999).
[l] Vectra, Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask, Revision 2, Docket No. 71-9255, February 1996 (TW 1999).
[m] NAC International, Inc., Safety Analysis Report for the UMS™ [Universal MPC System] Universal Transport Cask, EA790-SAR-001, Docket No. 71-9270, Revision 1, June 1999 (NAC 1999a).
[n] NAC International, Inc., NAC-STC SAR docket No. 71-9235, Revision 10, February 1999 (NAC 1999b).
[o] Transportation Cask Physical Envelope Study Report (CRWMS M&O 1998a).
[p] Cask Fleet Costs for the 1999 Total System Life Cycle Cost Report Update (CRWMS M&O 1999b).

D.1.2 Cask Handling Interfaces

Information to be developed for subsequent revision.

D.1.3 Cask Closure Interfaces

Information to be developed for subsequent revision.

D.2 PROJECTED SEASONALITY OF SHIPMENTS

Information to be developed for subsequent revision.

D.3 ADDITIONAL DESIGN CONSIDERATIONS

Information to be developed for subsequent revision.

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APPENDIX E

DESIGN BASIS WASTE INPUT RESULTS

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DESIGN BASIS WASTE INPUT RESULTS

This appendix contains detailed waste stream information (chosen from the cases reviewed), that provides the most limiting of all the results. Estimated heat and burnup distributions for the BWR and PWR fuel upon arrival at the MGR are provided in Tables E-1 and E-2. Transportation cask arrivals are displayed in Tables E-3 through E-6.

This data is based on specific fuel selection assumptions. The tables are not based on annual arrival data. The percentage of fuel in a particular heat bin, upon arrival, falls within a burnup range for each fuel type, (e.g., 16 percent of the BWR fuel has a heat of 100-149 watts at the time of arrival, and 7.8 percent of the BWR fuel has a heat of 100-149 watts and falls within the burnup range of 30,000 to 34,999 MWd/MTU at arrival).

Tables E-3 through E-6 display transportation cask arrivals sorted in several different ways. Table E-3 shows the number of transportation casks that may arrive at the MGR receiving facility in any random year, sorting from the year that has the maximum number of legal weight truck casks to the minimum number. Note that this information provides guidance on the frequency at which any specific combination of casks may occur. Tables E-4, E-5, and E-6 display similar information based on sorting by the number of SPC rail casks, DPC rail casks, and the total number of shipments, respectively.

Table E-1. Heat Distribution for BWR Fuel

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies[a]
0 - 49	0	4,999	1.22
	5,000	9,999	1.34
	10,000	14,999	2.25
	15,000	19,999	0.59
	20,000	24,999	0.17
	25,000	29,999	0.00
Total			5.57
50 - 99	5,000	9,999	0.02
	10,000	14,999	3.96
	15,000	19,999	5.09
	20,000	24,999	7.28
	25,000	29,999	7.80
	30,000	34,999	0.42
Total			24.58
100 - 149	15,000	19,999	0.26
	20,000	24,999	1.38
	25,000	29,999	4.22
	30,000	34,999	7.77
	35,000	39,999	2.35
Total			15.98
150 - 199	20,000	24,999	0.07
	25,000	29,999	1.44
	30,000	34,999	4.68
	35,000	39,999	4.03
	40,000	44,999	0.64
	45,000	49,999	0.05
Total			10.92
200 - 249	25,000	29,999	0.09
	30,000	34,999	2.10
	35,000	39,999	4.23
	40,000	44,999	2.02
	45,000	49,999	0.37
	50,000	54,999	0.07
Total			8.88

Table E-1. Heat Distribution for BWR Fuel (Continued)

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies [a]
250 - 299	30,000	34,999	0.15
	35,000	39,999	4.10
	40,000	44,999	3.42
	45,000	49,999	1.83
	50,000	54,999	0.72
	55,000	59,999	0.25
	60,000	64,999	0.03
Total			10.50
300 - 349	35,000	39,999	0.99
	40,000	44,999	6.42
	45,000	49,999	4.40
	50,000	54,999	1.16
	55,000	59,999	0.31
	60,000	64,999	0.05
	65,000	69,999	0.01
Total			13.33
350 - 399	30,000	34,999	0.01
	35,000	39,999	0.06
	40,000	44,999	0.55
	45,000	49,999	2.85
	50,000	54,999	3.06
	55,000	59,999	1.17
	60,000	64,999	0.41
	65,000	69,999	0.01
Total			8.12
400 - 449	35,000	39,999	0.05
	40,000	44,999	0.16
	45,000	49,999	0.37
	50,000	54,999	0.36
	55,000	59,999	0.24
	60,000	64,999	0.11
	65,000	69,999	0.13
Total			1.42

Table E-1. Heat Distribution for BWR Fuel (Continued)

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies [a]
450 - 499	35,000	39,999	0.10
	40,000	44,999	0.09
	45,000	49,999	0.11
	50,000	54,999	0.20
	55,000	59,999	0.06
Total			**0.55**
500 - 549	40,000	44,999	0.02
	45,000	49,999	0.05
	50,000	54,999	0.05
	55,000	59,999	0.06
	60,000	64,999	0.01
Total			**0.18**
Grand Total			**100.00**

Note: [a] Column may not add to 100 percent due to rounding.
[b] Based on Origin 2 Heat Code.

Table E-2. Heat Distribution for PWR Fuel

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies [b]
0 - 99	0	4,999	0.03
	5,000	9,999	0.20
	10,000	14,999	0.59
	15,000	19,999	0.08
	20,000	24,999	0.06
	25,000	29,999	0.04
Total			1.01
100 - 199	10,000	14,999	0.83
	15,000	19,999	2.72
	20,000	24,999	1.62
	25,000	29,999	2.46
	30,000	34,999	0.64
	45,000	49,999	0.00
Total			8.26
200 - 299	10,000	14,999	0.13
	15,000	19,999	1.67
	20,000	24,999	0.71
	25,000	29,999	3.31
	30,000	34,999	7.15
	35,000	39,999	4.41
	40,000	44,999	0.54
	45,000	49,999	0.00
Total			17.92
300 - 399	10,000	14,999	0.01
	15,000	19,999	0.39
	20,000	24,999	0.33
	25,000	29,999	0.29
	30,000	34,999	1.87
	35,000	39,999	6.28
	40,000	44,999	4.68
	45,000	49,999	0.96
	50,000	54,999	0.00
Total			14.81

Table E-2. Heat Distribution for PWR Fuel (Continued)

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies [b]
400 - 499	15,000	19,999	0.08
	20,000	24,999	0.06
	25,000	29,999	0.09
	30,000	34,999	0.64
	35,000	39,999	1.88
	40,000	44,999	3.69
	45,000	49,999	1.96
	50,000	54,999	0.37
	55,000	59,999	0.03
Total			8.80
500 - 599	20,000	24,999	0.02
	25,000	29,999	0.04
	30,000	34,999	0.22
	35,000	39,999	1.50
	40,000	44,999	2.83
	45,000	49,999	2.56
	50,000	54,999	0.61
	55,000	59,999	0.11
	60,000	64,999	0.02
Total			7.91
600 - 699	25,000	29,999	0.01
	30,000	34,999	0.13
	35,000	39,999	0.57
	40,000	44,999	2.38
	45,000	49,999	2.92
	50,000	54,999	1.30
	55,000	59,999	0.24
	60,000	64,999	0.02
Total			7.57
700 - 799	25,000	29,999	0.00
	30,000	34,999	0.05
	35,000	39,999	0.23
	40,000	44,999	1.60
	45,000	49,999	3.55
	50,000	54,999	2.39
	55,000	59,999	0.72

Table E-2. Heat Distribution for PWR Fuel (Continued)

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies [b]
700 - 799	60,000	64,999	0.08
	65,000	69,999	0.01
Total			8.61
800 - 999	30,000	34,999	0.02
	35,000	39,999	0.14
	40,000	44,999	0.49
	45,000	49,999	2.71
	50,000	54,999	6.09
	55,000	59,999	3.72
	60,000	64,999	1.06
	65,000	69,999	0.05
Total			14.27
1000 - 1199	35,000	39,999	0.03
	40,000	44,999	0.09
	45,000	49,999	0.34
	50,000	54,999	0.87
	55,000	59,999	2.50
	60,000	64,999	2.41
	65,000	69,999	0.31
Total			6.54
1200 - 1399	40,000	44,999	0.01
	45,000	49,999	0.07
	50,000	54,999	0.19
	55,000	59,999	0.37
	60,000	64,999	0.78
	65,000	69,999	0.53
	70,000	74,999	0.02
Total			1.97
1400 - 1599	50,000	54,999	0.04
	55,000	59,999	0.13
	60,000	64,999	0.30
	65,000	69,999	0.06
	70,000	74,999	0.02
Total			0.55

Table E-2. Heat Distribution for PWR Fuel (Continued)

Heat Range (watts)	Min Burnup (MWd/MTU)	Max Burnup (MWd/MTU)	Percent of Assemblies [b]
1600 - 1799	50,000	54,999	0.01
	55,000	59,999	0.03
	60,000	64,999	0.16
	65,000	69,999	0.03
	70,000	74,999	0.00
Total			0.24
1800 - 1999	60,000	64,999	0.02
	65,000	69,999	0.08
Total			0.10
MOX [a]	15,000	19,999	0.04
	20,000	24,999	0.08
	30,000	34,999	0.01
	35,000	39,999	0.02
	40,000	44,999	0.42
	45,000	49,999	0.40
	50,000	54,999	0.25
	55,000	59,999	0.14
	60,000	64,999	0.07
	65,000	69,999	0.01
Total			1.44
Grand Total			100.00

NOTE: [a] Heat is not calculated for MOX fuel.
[b] Column may not add to 100 percent due to rounding.
[c] Based on Origin 2 Heat Code.

Table E-3. Design Basis CSNF Shipment Arrivals - Sorted by Truck Cask Arrivals

	Truck [a]	UCF Rail [a]	DPC Rail [a]	Total [a]
	230	350	40	620
	220	340	40	580
	180	210	40	420
	150	380	40	560
	120	380	40	540
	100	380	50	530
	100	140	20	250
	90	340	70	490
	70	380	50	490
	60	80	20	140
	50	290	110	430
	40	330	90	450
	40	360	80	460
	30	40	20	80
	30	330	80	430
	30	310	100	430
	30	300	110	430
	30	350	80	450
	0	330	90	410
	0	240	150	380
	0	230	140	370
	0	220	160	380
	0	140	210	340
	0	110	220	330
	0	100	240	340
	0	70	240	310
	0	60	260	320
	0	50	250	300
	0	60	250	310
	0	30	260	290
	0	40	140	170
	0	0	0	0
Total	1,510	6,840	3,550	11,900
Maximum	230	380	260	620

NOTE: [a] Numbers have been rounded up to the nearest 10 casks.
[b] Totals do not add due to rounding.

Table E-4. Design Basis CSNF Shipment Arrivals - Sorted by SPC Rail Cask Arrivals

	Truck [a]	UCF Rail [a]	DPC Rail [a]	Total [a]
	150	380	40	560
	120	380	40	540
	100	380	50	530
	70	380	50	490
	40	360	80	460
	230	350	40	620
	30	350	80	450
	90	340	70	490
	220	340	40	580
	40	330	90	450
	30	330	80	430
	0	330	90	410
	30	310	100	430
	30	300	110	430
	50	290	110	430
	0	240	150	380
	0	230	140	370
	0	220	160	380
	180	210	40	420
	100	140	20	250
	0	140	210	340
	0	110	220	330
	0	100	240	340
	60	80	20	140
	0	70	240	310
	0	60	260	320
	0	60	250	310
	0	50	250	300
	30	40	20	80
	0	40	140	170
	0	30	260	290
	0	0	0	0
Total	1,510	6,840	3,550	11,900
Maximum	230	380	260	620

NOTE: [a] Numbers have been rounded up to the nearest 10 casks.
[b] Totals do not add due to rounding.

Table E-5. Design Basis CSNF Shipment Arrivals - Sorted by DPC Rail Cask Arrivals

	Truck[a]	UCF Rail[a]	DPC Rail[a]	Total[a]
	0	60	260	320
	0	30	260	290
	0	60	250	310
	0	50	250	300
	0	70	240	310
	0	100	240	340
	0	110	220	330
	0	140	210	340
	0	220	160	380
	0	240	150	380
	0	230	140	370
	0	40	140	170
	30	300	110	430
	50	290	110	430
	30	310	100	430
	40	330	90	450
	0	330	90	410
	30	330	80	430
	30	350	80	450
	40	360	80	460
	90	340	70	490
	100	380	50	530
	70	380	50	490
	180	210	40	420
	230	350	40	620
	120	380	40	540
	220	340	40	580
	150	380	40	560
	100	140	20	250
	30	40	20	80
	60	80	20	140
	0	0	0	0
Total	1,510	6,840	3,550	11,900
Maximum	230	380	260	620

NOTE: [a] Numbers have been rounded up to the nearest 10 casks.
[b] Totals do not add due to rounding.

Table E-6. Design Basis CSNF Shipment Arrivals - Sorted by Total Cask Arrivals

	Truck [a]	UCF Rail [a]	DPC Rail [a]	Total [a]
	230	350	40	620
	220	340	40	580
	150	380	40	560
	120	380	40	540
	100	380	50	530
	90	340	70	490
	70	380	50	490
	40	360	80	460
	40	330	90	450
	30	350	80	450
	30	330	80	430
	50	290	110	430
	30	300	110	430
	30	310	100	430
	180	210	40	420
	0	330	90	410
	0	240	150	380
	0	220	160	380
	0	230	140	370
	0	140	210	340
	0	100	240	340
	0	110	220	330
	0	60	260	320
	0	70	240	310
	0	60	250	310
	0	50	250	300
	0	30	260	290
	100	140	20	250
	0	40	140	170
	60	80	20	140
	30	40	20	80
	0	0	0	0
Total	1,510	6,840	3,550	11,900
Maximum	230	380	260	620

NOTE: [a] Numbers have been rounded up to the nearest 10 casks.
[b] Totals do not add due to rounding.

APPENDIX F

GENERAL FUEL SPECIFICATIONS

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GENERAL FUEL SPECIFICATIONS

The following text, Appendix E of 10 CFR 961 provides the specifications for standard fuel and is included here for ease of reference:

A. *Fuel Category Identification*

1. Categories—Purchaser shall use reasonable efforts, utilizing technology equivalent to and consistent with the commercial practice, to properly classify Spent Nuclear Fuel (SNF) prior to delivery to DOE, as follows:
   a. *Standard Fuel* means SNF that meets all the General Specifications therefor set forth in paragraph B below.
   b. *Nonstandard Fuel* means SNF that does not meet one or more of the General Specifications set forth in subparagraphs 1 through 5 of paragraph B below, and which is classified as Nonstandard Fuel Classes NS–1 through NS–5, pursuant to paragraph B below.
   c. *Failed Fuel* means SNF that meets the specifications set forth in subparagraphs 1 through 3 of paragraph B below, and which is classified as Failed Fuel Class F–1 through F–3 pursuant to subparagraph 6 of paragraph B below.
   d. Fuel may have "Failed Fuel" and/or several "Nonstandard Fuel" classifications

B. *Fuel Description and Subclassification—General Specifications*

1. *Maximum Nominal Physical Dimensions.*

	Boiling water reactor (BWR)	Pressurized water reactor (PWR)
Overall Length	14 feet, 11 inches.	14 feet, 10 inches.
Active Fuel Length	12 feet, 6 inches	12 feet, 0 inches.
Cross Section[1]	6 inches x 6 inches.	9 inches x 9 inches.

[1] The cross section of the fuel assembly shall not include the channel.
NOTE: Fuel that does not meet these specifications shall be classified as Nonstandard Fuel—Class NS–1.

2. *Nonfuel Components.* Nonfuel components including, but not limited to, control spiders, burnable poison rod assemblies, control rod elements, thimble plugs, fission chambers, and primary and secondary neutron sources, that are contained within the fuel assembly, or BWR channels that are an integral part of the fuel assembly, which do not require special handling, may be included as part of the spent nuclear fuel delivered for disposal pursuant to this contract.

   NOTE: Fuel that does not meet these specifications shall be classified as Nonstandard Fuel—Class NS–2.

3. *Cooling*. The minimum cooling time for fuel is five (5) years.
   NOTE: Fuel that does not meet this specification shall be classified as Nonstandard Fuel—Class NS–3.

4. *Non-LWR Fuel.* Fuel from other than LWR power facilities shall be classified as Nonstandard Fuel—Class NS-4. Such fuel may be unique and require special handling, storage, and disposal facilities.

5. *Consolidated Fuel Rods*. Fuel, which has been disassembled and stored with the fuel rods in a consolidated manner shall be classified as Nonstandard Fuel Class NS–5.

6. *Failed Fuel.*
   a. Visual Inspection.
      Assemblies shall be visually inspected for evidence of structural deformity or damage to cladding or spacers which may require special handling. Assemblies which [i] are structurally deformed or have damaged cladding to the extent that special handling may be required or [ii] for any reason cannot be handled with normal fuel handling equipment shall be classified as Failed Fuel—Class F–1.
   b. Previously Encapsulated Assemblies.
      Assemblies encapsulated by Purchaser prior to classification hereunder shall be classified as Failed Fuel—Class F–3. Purchaser shall advise DOE of the reason for the prior encapsulation of assemblies in sufficient detail so that DOE may plan for appropriate subsequent handling.
   c. Regulatory Requirements.
      Spent fuel assemblies shall be packaged and placed in casks so that all applicable regulatory requirements are met.

C. Summary of Fuel Classifications

1. *Standard Fuel*:
   a. Class S–1: PWR
   b. Class S–2: BWR

2. *Nonstandard Fuel*:
   a. Class NS–1: Physical Dimensions
   b. Class NS–2: Non Fuel Components
   c. Class NS–3: Short Cooled
   d. Class NS–4: Non-LWR
   e. Class NS–5: Consolidated Fuel Rods.

3. *Failed Fuel*:
   a. Class F–1: Visual Failure or Damage
   b. Class F–2: Radioactive "Leakage"
   c. Class F–3: Encapsulated

D. High-Level Radioactive Waste

The DOE shall accept high-level radioactive waste. Detailed acceptance criteria and general specifications for such waste will be issued by the DOE no later than the date on which DOE submits its license application to the Nuclear Regulatory Commission for the first disposal facility.

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