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I. Listing of Computer Files	7
II. One Compact Disc Containing All Files Listed in Attachment I (1 of 1)	0
III. Draft Sketches of FHF Moderator Control Areas	10

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LIST OF ACRONYMS AND ABBREVIATIONS

BSC	Bechtel SAIC Company, LLC
BWR	Boiling Water Reactor
CD	Compact Disc
CFR	Code of Federal Regulations
cm	centimeters
CRWMS	Civilian Radioactive Waste Management System
CSNF	Commercial Spent Nuclear Fuel
DOE	U.S. Department of Energy
DPC	Dual-Purpose Canister
FFTF	Fast Flux Test Facility
FHF	Fuel Handling Facility
HLW	High-Level Waste
i.d.	inner diameter
k_{eff}	neutron effective multiplication factor
kg	kilograms
LWBR	Light Water Breeder Reactor
MCNP	Monte Carlo N-Particle transport code
MCO	Multi Canister Overpack
MGR	Monitored Geological Repository
MPC	Multi Purpose Canister
MSC	MGR Site specific Cask
NRC	U.S. Nuclear Regulatory Commission
o.d.	outer diameter
OFA	Optimized Fuel Assembly
PDC	Project Design Criteria
PWR	Pressurized Water Reactor
SS	Stainless Steel
SNF	Spent Nuclear Fuel
TMI	Three Mile Island
TRIGA	Training Research Isotopes General Atomics
WP	Waste Package
wt %	weight percent

1. PURPOSE

The purpose of this design calculation is to perform a criticality evaluation of the Fuel Handling Facility (FHF) and the operations and processes performed therein. The current intent of the FHF is to receive transportation casks whose contents will be unloaded and transferred to waste packages (WP) or MGR Specific Casks (MSC) in the fuel transfer bays. Further, the WPs will also be prepared in the FHF for transfer to the sub-surface facility (for disposal). The MSCs will be transferred to the Aging Facility for storage. The criticality evaluation of the FHF features the following:

- Consider the types of waste to be received in the FHF as specified below:

    1. Uncanistered commercial spent nuclear fuel (CSNF)
    2. Canistered CSNF (with the exception of horizontal dual-purpose canister (DPC) and/or multi-purpose canisters (MPCs)
    3. Navy canistered SNF (long and short)
    4. Department of Energy (DOE) canistered high-level waste (HLW)
    5. DOE canistered SNF (with the exception of MCOs)

- Evaluate the criticality analyses previously performed for the existing Nuclear Regulatory Commission (NRC)-certified transportation casks (under 10 CFR 71) to be received in the FHF to ensure that these analyses address all FHF conditions including normal operations, and Category 1 and 2 event sequences.

- Evaluate FHF criticality conditions resulting from various Category 1 and 2 event sequences. Note that there are currently no Category 1 and 2 event sequences identified for FHF. Consequently, potential hazards from a criticality point of view will be considered as identified in the *Internal Hazards Analysis for License Application* document (BSC 2004c, Section 6.6.4).

- Assess effects of potential moderator intrusion into the fuel transfer bay for defense in depth.

The SNF/HLW waste transfer activity (i.e., assembly and canister transfer) that is being carried out in the FHF has been classified as safety category in the *Q-list* (BSC 2003, p. A-6). Therefore, this design calculation is subject to the requirements of the *Quality Assurance Requirements and Description* (DOE 2004), even though the FHF itself has not yet been classified in the Q-list. Performance of the work scope as described and development of the associated technical product conform to the procedure AP-3.12Q, *Design Calculations and Analyses*.

It should also be mentioned that the facility description document or the system description document for the FHF is not available at this time. Consequently, this calculation is valid for the current design and may not reflect the ongoing design evolution of the FHF.

2. METHOD

2.1 CRITICALITY SAFETY ANALYSIS

The criticality safety calculations presented in this document evaluate the various waste forms in the transportation casks, WPs, canisters, and MSCs casks in the FHF to ensure they all meet the criticality safety requirements under normal conditions as well as for Category 1 and 2 events. Moderator conditions are also varied to find the most reactive configuration. The process and methodology for criticality safety analysis given in the *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7) will be implemented in these calculations. The following method will be pursued for each waste form and cask/canister configuration (BSC 2004e, Section 2.2.7):

- The design basis for the FHF relies on the most reactive fuel assemblies
- The multiplication factor (k_{eff}) will not exceed 0.95, including all biases and uncertainties in the data and method of the analysis, under all normal, and Category 1 and 2 event sequences
- Conservative modeling dimensional variables will be used (e.g., assembly pitch, manufacturing tolerances for assemblies, etc.) in order to maximize reactivity
- Conservative modeling assumptions will also be used regarding materials in fuel including no accounting for burnable poisons in fuel, no credit for ^{234}U and ^{236}U in fuel, and use of the most reactive fuel stack density
- Credit can only be taken for up to 75 % of the neutron absorbing material in criticality controls (e.g., grid plates).
- Moderator density will be varied over the range of 0.0 through 1.0 in order to include all possible criticality conditions.

In addition, the effect on criticality due to rearrangement of fuel inside the casks/canisters from natural events (e.g., earthquakes) or incidents (e.g., dropped assembly) will be considered (BSC 2004e, Section 2.2.8).

These calculations use the qualified software MCNP (Briesmeister 1997 and CRWMS M&O 1998a). MCNP is a three-dimensional Monte Carlo particle transportation code with the capability to calculate eigenvalues for critical systems. The Nuclear Regulatory Commission (NRC) accepts MCNP in NUREG-1567 (NRC 2000, p. 8-10) for criticality calculations.

2.2 ELECTRONIC MANAGEMENT OF INFORMATION

Electronic management of information generated from these calculations is controlled in accordance with AP-3.13Q, *Design Control*. The computer input and output files generated from this calculation are stored on a Compact Disc (CD), and submitted as an attachment to this document (Attachment II).

3. ASSUMPTIONS

3.1 The nominal acceptable calculated value of k_{eff} is assumed to be 0.925 as a criticality limit in order to meet the design criteria specified in the PDC Document [i.e., k_{eff} can not exceed 0.95 including uncertainties and bias at 95% confidence level (Minwalla 2003, Section 4.9.2.2.1)]. In other words, the nominal value provides a margin of 0.025 (0.95 - 0.925) to account for code bias and uncertainties at 95% confidence level.

Rationale: Uncertainties and bias that need to be considered in this analysis pertain to statistical uncertainties, dimensional uncertainties, code bias, and tolerance uncertainties. Applicable code bias for the fuel type and enrichment range of this analysis is typically less than 0.5 % (CRWMS M&O 1999c, Section 4). An allowance of 2% is provided to account for the remaining uncertainties associated with statistical variation, dimensional variables and tolerances. This allowance is similar to, and slightly greater than (conservative), the value used for the SNF storage and transportation cask criticality evaluations (General Atomics 1993, p. 6.4-7).

Usage: This assumption is used throughout this design calculation.

3.2 The naval waste packages will be designed in such a way as to make criticality not credible.

Rationale: Criticality analyses for naval waste packages are the responsibility of the U.S Department of the Navy. Per the *Naval Spent Nuclear Fuel Waste Package System Description* document, the "sealed waste package shall provide criticality control" to reduce the probability of a criticality occurring (BSC 2004d, Section 3.1.1.3).

Usage: Sections 5.1 and 5.2.3.

3.3 The MGR Site specific Cask (MSC) is assumed to be similar in design, other than the neutron poison loading/configuration, to the Multi Purpose Canister (MPC)-24 for PWR fuel and the MPC-68 for BWR fuel.

Rationale: Since the MSC is still being developed, the criticality control features will be similar to the existing NRC-certified storage casks.

Usage: This assumption is used in Sections 5.1, 5.2.3, and 6.1.

3.4 The Fort St. Vrain fuel is assumed to have a U-235 enrichment of 100%.

Rationale: This assumption was used to introduce conservatism into the calculation.

Usage: Section 5.1.

3.5 It is assumed that the fuel basket inside the canister (if present) remains intact following a canister drop.

Rationale: The canister internals are designed to remain sufficiently intact that there would be no criticality concern following any credible drop or handling mishap (Minwalla 2003, Section 4.9.2.2.7).

Usage: Section 6.2.

3.6 For damaged fuel calculations, it is assumed that the inside of the canister is dry.

Rationale: The canister will be shipped to the repository dry. There is no credible mechanism by which the inside of the canister could become flooded since the canister is designed to withstand any credible drop without breaching (Canori and Leitner 2003, p. 3-63).

Usage: Section 6.2.

4. USE OF COMPUTER SOFTWARE

4.1 BASELINED SOFTWARE

4.1.1 MCNP

The MCNP code (CRWMS M&O 1998a) was used to calculate the multiplication factor, k_{eff}, for all systems presented in this report. The software specifications are as follows:

- Program Name: MCNP (CRWMS M&O 1998a)
- Version/Revision Number: Version 4B2LV
- Status/Operating System: Qualified/HP-UX B.10.20
- Software Tracking Number: 30033 V4B2LV
- Computer Type: HP 9000 Series Workstations
- CPU Number: 700887

The input and output files for the various MCNP calculations are contained on a CD (Attachment II) and the files are listed in Attachment I.

The MCNP software used was: (1) appropriate for the criticality (k_{eff}) calculations, (2) used only within the range of validation as documented through Briesmeister (1997) and CRWMS M&O (1998b, Section 3.1), and (3) obtained from Software Configuration Management in accordance with appropriate procedures.

4.2 COMMERCIAL OFF-THE-SHELF SOFTWARE

4.2.1 MICROSOFT EXCEL 97 SR-2

- Title: Excel
- Version/Revision Number: Microsoft® Excel 97 SR-2
- This version is installed on a PC running Microsoft Windows 2000 with CPU number 503009

The files for the various Excel calculations are contained on a CD (Attachment II) and the files are listed in Attachment I.

The Excel software was used to illustrate results in Sections 5.2 and 6. Excel is exempt from qualification per Section 2.1.6 of LP-SI.11Q, *Software Management*.

5. CALCULATION

All technical product inputs and sources of the inputs used in the development of this calculation are documented in this section. It should also be mentioned that the terms "model(s)" and "modeling" as used in this calculation document refer to the geometric configurations of the criticality cases analyzed.

5.1 CALCULATIONAL INPUTS

5.1.1 Design Requirements and Criteria

The design criteria for criticality safety analysis provided in Section 4.9.2.2 of the *Project Design Criteria* document (Minwalla 2003) are used in these calculations. The pertinent criteria for surface facility criticality include the following (Minwalla 2003, Section 4.9.2.2):

- The multiplication factor (k_{eff}) will not exceed 0.95, including all biases and uncertainties in the data and method of the analysis, under all normal and off-normal event sequences. This criterion satisfies Requirement Number PRD-013/T-022 in the *Project Requirements Document* (Canori and Leitner 2003, p. 3-76).

- The facility design will utilize a favorable geometry and/or fixed neutron absorbers without the use of burnup credit.

- No moderator shall be present in any area where radioactive waste is being handled (fuel transfer rooms, WP closure room etc.). Attachment III features draft sketches of the FHF as of the date of this calculation (the general arrangement drawings have not been finalized yet), and may not reflect the ongoing design evolution. The purpose of the sketches is to show functional areas where moderator control is engineered in the design for criticality safety. These functional areas will remain with moderator control, even if design changes are made to the FHF with respect to the layout.

5.1.2 Most Reactive Fuel Selection

In accordance with the requirements given in *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7), the design of the facility should be based on most reactive fuel assemblies. The following evaluations were performed with MCNP to determine the most reactive fuel assembly.

<u>Commercial Fuel</u>

The evaluations performed in the *Surface Facility Criticality Safety Calculations* document show that the W 17 x 17 OFA was found to be the most reactive PWR fuel assembly when modeled as a single fuel assembly and in a storage cask configuration (BSC 2004f, Section 5.2.1). In addition, an evaluation to determine the most reactive fuel assembly was performed in the *Final Safety and Analysis Report for the Holtec International Storage and Transfer Operation*

Reinforced Module Cask System (HI-STORM 100 Cask System) and the Westinghouse 17x17 OFA was selected (Holtec International 2002, Section 6.2-2). The GE 8 x 8 array was selected for the BWR fuel (Holtec International 2002, Section 6.2-3 as featured in the *Aging Facility Criticality Safety Calculations* document (BSC 2004a, Section 5.1.3). The results from the *Surface Facility Criticality Safety Calculations* and *Aging Facility Criticality Safety Calculation* documents will be utilized later in this FHF evaluation where applicable.

DOE Fuel

DOE SNF has been categorized into nine fuel groups (Mecham, D.C. 2004, Section 4.2.4.1):

1. Uranium Metal fuels (N-Reactor)
2. Uranium-Zirconium/Uranium-Molybdenum fuels (Enrico Fermi Liquid Metal Reactor)
3. Uranium Oxide fuels (high enriched uranium - Shippingport PWR)
4. Uranium Oxide fuels (low enriched uranium - Three Mile Island (TMI)-2 PWR)
5. Uranium-Aluminum fuels (foreign research reactor – Melt & Dilute)
6. Uranium/Thorium/Plutonium Carbide fuels (Ft. St. Vrain Gas Cooled Reactor)
7. Mixed Oxide fuels (Fast Flux Test Facility (FFTF) Reactor)
8. Uranium/Thorium Oxide fuels (Shippingport Light Water Breeder Reactor (LWBR))
9. Uranium-Zirconium-Hydride fuels (Training Research Isotopes General Atomics (TRIGA)).

The *Canister Handling Criticality Safety Calculations* document evaluates the DOE fuel types listed above (both Mark 1A and Mark IV type fuel are considered for N Reactor and type "D" and type "K" canister are evaluated for TMI-2 fuel) (BSC 2004b, Section 5.1.2). The FFTF fuel type was shown to be the most reactive in a single flooded canister under normal conditions (BSC 2004b, Section 6.1). However, during off-normal conditions the TRIGA fuel is the more reactive fuel type (BSC 2004b, Section 6.2). Additional calculations will be presented later in this document for all DOE fuel types to show the effect of varying moderator density (Section 6.2.1) and effects of neutron poison variation for applicable DOE fuel types (Section 6.2.2). An off-normal condition with TRIGA fuel will also be shown later in this document (Section 5.2).

Naval Fuel Canisters

The design and safety analyses of the naval spent nuclear fuel WP are the responsibility of the U. S. Department of the Navy. No data on the naval waste package, including criticality analyses, are publicly available. Section 3.1.1.3 of *Naval Spent Nuclear Fuel Waste Package System Description Document* (BSC 2004d) states "The sealed waste package shall provide criticality control ... The term 'control' used in this requirement is intended to mean that neutron absorber materials are added and the waste package loading configuration design reduces the probability of a criticality occurring." From this description, it is assumed that the naval canister and internals will be designed to make criticality not credible (Assumption 3.2) and no evaluations will be featured in this document.

5.1.3 Upper Subcritical Limit

In accordance with the requirements given in *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7), k_{eff} should not exceed 0.95, including all biases and uncertainties in the data and method of the analysis. All evaluations featuring a cask and the MSC are performed for the worst case combination of manufacturing tolerances with respect to criticality (Holtec International 2002, p.6.3-2). Evaluations were performed to determine the effects of tolerances (Holtec International 2002, Tables 6.3-1 & 6.3-2). It was determined that design parameters important to criticality safety are fuel enrichment, the inherent geometry of the fuel basket structure and the fixed neutron absorbing panels (Boral) (Holtec International 2002, p. 6.3-3). Further, the results referred to in Section 6 of this report are within the bounds of the k_{eff} values demonstrated in the *Final Safety and Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System)* to cover uncertainties and bias. The remainder of the results either presented or referred to in this document is designed to meet an upper subcritical limit of 0.925 per Assumption 3.1.

5.1.4 Geometry Calculation Inputs

Physical inputs for the various casks/canisters and waste forms are described in this section. Since criticality evaluations have been performed for some of the casks/canisters and waste forms previously, the inputs will be referenced from these previous studies.

<u>Commercial Fuel (Transportation Casks and MSCs)</u>

A representative vertical cask is selected here for criticality calculations to demonstrate compliance with the criticality safety requirements. The selected cask is HI-STORM 100, as this system is currently qualified for high seismic requirements to ensure that the YMP seismic spectrum will be enveloped (Cogema 2004, p.5). Since the MSC is still being developed, it was assumed that it is similar in design to the MPC-24 of the HI-STORM 100 cask system for PWR fuel and MPC-68 of the HI-STORM 100 cask system for BWR fuel (Assumption 3.3). A criticality evaluation for these casks has already been performed in the *Aging Facility Criticality Safety Calculations* document, where the configuration and physical dimensions for the MPC-24 as well as fuel specifications (BSC 2004a, Section 5.1.5.1) and the MPC-68 (BSC 2004a, Section 5.1.5.3) are given. Figure 5.1-1 illustrates the radial view of the MPC-24 cask inside the HI-STORM 100 overpack and Figure 5.1-2 illustrates the radial view of the MPC-68 cask inside the HI-STORM 100 overpack. Figure 5.1-2 also displays the axial view of the MPC inside the overpack, which configuration is the same for both the MPC-24 and MPC-68 casks.

It should be mentioned that calculations of the MSCs only exist for commercial SNF, which was valid at the time of the criticality safety calculation performed for the Aging Facility. This criticality safety calculation is in the process of being revised to include other waste types for aging.

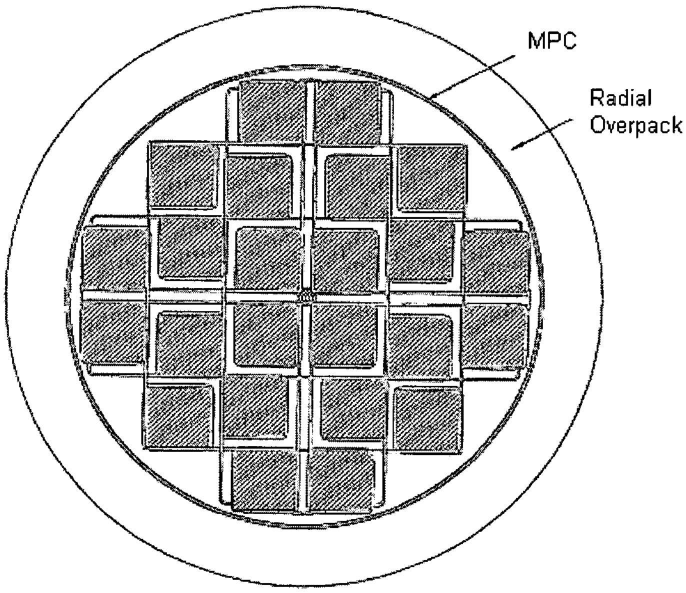

Figure 5.1-1 Radial View of the MPC-24 Cask Inside the HI-STORM 100 Overpack

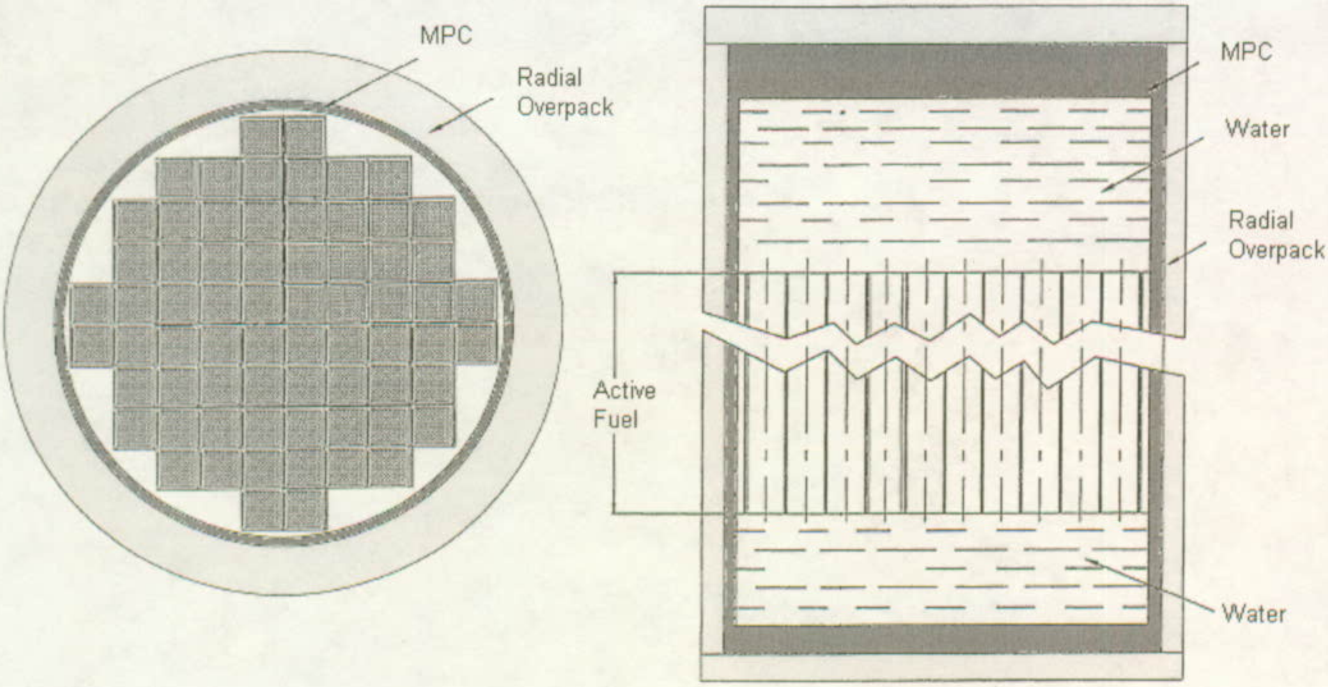

Figure 5.1-2 Radial and Axial View of the MPC-68 Cask Inside the HI-STORM 100 Overpack

DOE Fuel (Canisters)

Table 5.1-1 presents the physical dimensions of the canisters and Table 5.1-2 shows the DOE fuel parameters. Figure 5.1-3 displays the DOE canisters considered in this evaluation, as described in Section 5.1.2, in the radial view. An axial representation of the DOE SNF canisters is also included in Figure 5.1-3. It should be mentioned that the MCNP input files from the *Canister Handling Criticality Safety Calculations* document (BSC 2004b) were used as a starting point for the calculations presented in this document. For more details regarding canister physical dimensions, see Section 5.1.4 (BSC 2004b) and Section 5.1.2 (BSC 2004b) for more specifics regarding DOE fuel parameters.

Table 5.1-1 Physical Dimensions of DOE Canisters

DOE Fuel Type	Canister o.d. (cm)	Canister length (cm)	Canister Capacity	Reference
Enrico Fermi	45.72		3360 fuel pins (2 sets of 12 tubes each containing 140 pins)	CRWMS M&O 2000a, p. 13
FFTF	45. 72 (0.95 cm wall thickness)	456.90 (414.50 cm internal length)	1302 fuel pins (6 assemblies with each 217 fuel pins)	CRWMS M&O 1999a, Figures 5-3 & 5-4
Fort St. Vrain	45. 72 (0.95 cm wall thickness)	457.0 (411.71 cm internal length)	5 fuel elements stacked vertically	BSC 2001a, p. 15
Melt & Dilute	45. 72 (0.95 cm wall thickness)	299.90 (254.0 cm internal length)	3 ingots stacked vertically	BSC 2001b, p.11
N Reactor	64.29	419.84	270 fuel elements (54 fuel elements stacked 5 high) [a]	CRWMS M&O 2001, p. 14 DOE 2000, pp. 23-25 (canister capacity)
Shippingport LWBR	45. 72 (0.95 cm wall thickness)	457.0 (411.71 cm internal length)	7428 fuel rods (12 assemblies with each 619 fuel rods)	CRWMS M&O 2000b, p. 18 DOE 1999b, p. 16 (canister capacity)
Shippingport PWR	45. 72 (0.95 cm wall thickness)	268.09 (internal length)	1 fuel cluster	CRWMS M&O 2000c, p. 15
TMI-2 (D canister) [b]	35. 56 (0.64 cm wall thickness)	380.37 (346.55 cm internal length)	1 fuel assembly (15x15 array having 204 fuel rods)	DOE 2003, pp. 21 (canister capacity), 25 & 26
TRIGA	45. 72 (0.95 cm wall thickness)	254.70 (internal length)	111 fuel elements (37 fuel elements stacked 3 high)	CRWMS M&O 1999d, p. 13

[a] Mark 1A contains 48 fuel elements stacked 5 high, comprising a total of 240 fuel elements (DOE 2000,Fig. 4-2).
[b] The K canister has a large internal diameter over which fuel matrix material is not constrained (see Fig. 5.1-3)

Table 5.1-2 DOE Fuel Parameters

DOE Fuel Type	Max. fissile enrichment (%)[a]	Fuel o.d. (cm)[b]	Clad i.d. (cm)	Clad o.d. (cm)	Pin Pitch (cm)[c]	Fuel length (cm)	Reference
Enrico Fermi	25.69	0.376	0.376	0.401	0.44	77.47	DOE 1999a, p.8 CRWMS M&O 2000a, p. 12 (clad)
FFTF	25.95	0.495	0.508	0.584	0.726	237.24	INEEL 2002, p.15, 17 (pin pitch) & Fig. 3 (fuel o.d.)
Fort St. Vrain	100.0 (Assumption 3.4)	1.245	–	–	1.880		Taylor 2001, p. 21 & Fig. 2-3 (pin pitch)
Melt & Dilute	20.0	41.91	–	–	–	76.2	BSC 2001c, p.3
N Reactor – outer fuel tube[d]	1.25[e]	6.096 4.496[f]	6.096 4.496	6.223 4.607	6.096	53.0	DOE 2000, Tables 3-1 & 3-2 (clad)
N Reactor – inner fuel tube[d]	1.25[e]	3.175 1.118[f]	3.175 1.118	3.378 1.245	6.096	53.0	DOE 2000, Tables 3-1 & 3-2 (clad)
Shippingport LWBR	4.90	0.640	0.734	0.778	0.937		DOE 1999b, p. 16 (enr.), Fig. 3-3 (pin pitch), Table 3-5 (fuel o.d.) & Table 3-8 (clad)
Shippingport PWR	93.2	–			–		DOE 1999c, Table 3-1
TMI-2	2.96	0.936	0.958	1.092	1.443	360.12	DOE 2003, p. 19 (enr.), p. 21, p. 22 (fuel length) & p.23 (fuel o.d.)
TRIGA	70.0	3.480 0.635[g]	3.490	3.592	3.480	38.10	CRWMS M&O 1999b, p. 19

[a] This is the total fissile content divided by the total heavy metal mass x 100.
[b] For fuel in the form of cylindrical rods, this is the fuel outside diameter
[c] For fuel in the form of cylindrical rods, this is the nominal pin pitch in the canister
[d] See Figure 5.1-3 for locations of outer and inner fuel tubes
[e] The enrichment for Mark IV (case B) is 0.95 %
[f] Inside diameters of fuel tubes
[g] Inside diameters of fuel tube

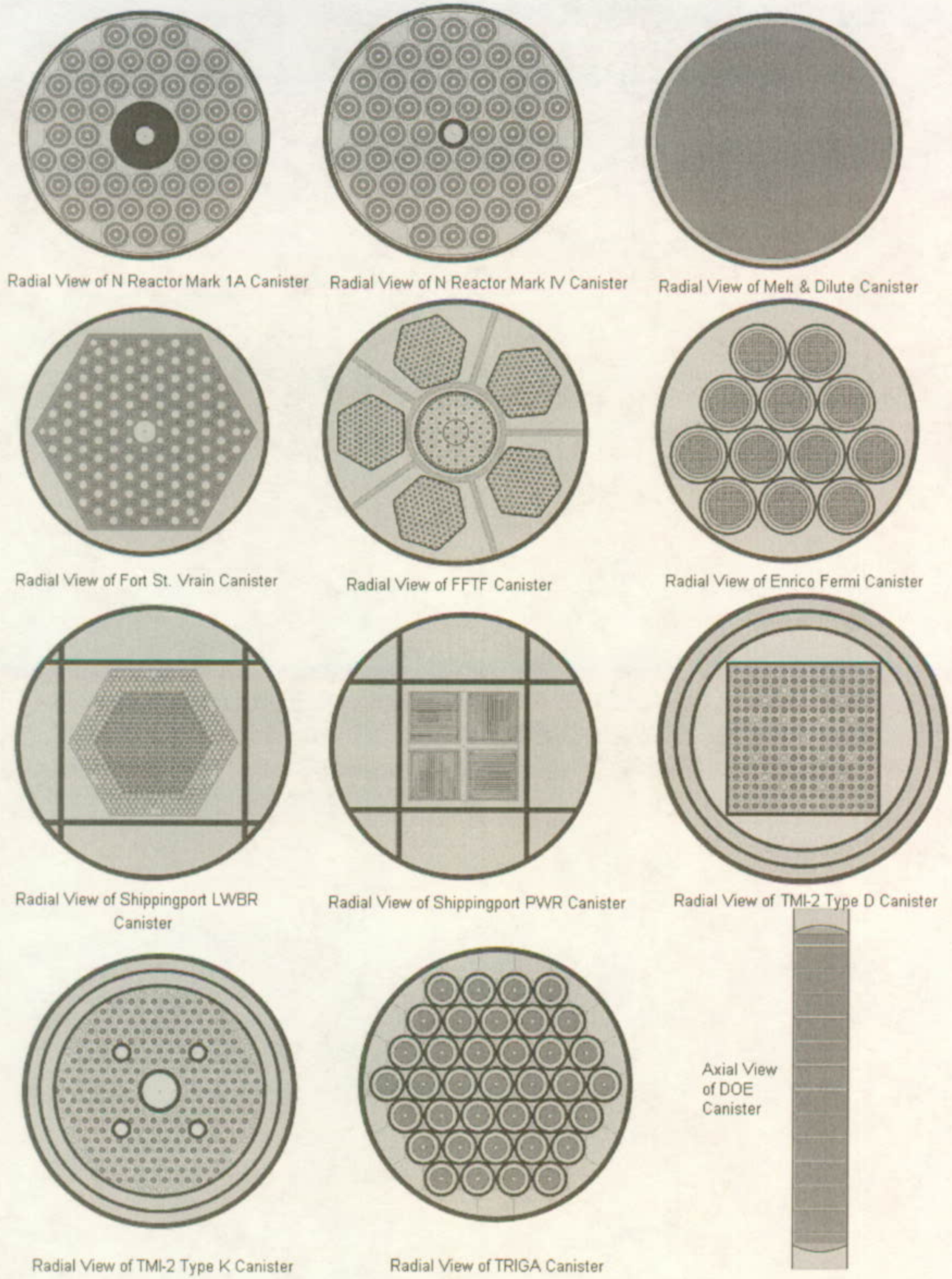

Figure 5.1-3 Radial and Axial View of the DOE Fuel Canisters

Navy Fuel

Per Assumption 3.2, no evaluations will be performed of navy fuel since it is the responsibility of the U. S. Department of the Navy.

5.1.5 Material Compositions

Material compositions for the various casks/canisters and waste forms are described in this section. Since criticality evaluations have been performed for some of the casks/canisters and waste forms previously, the inputs will be referenced from these previous studies.

Commercial Fuel (PWR & BWR)

Since no additional calculations were performed in this document for commercial fuel, the material compositions can be found in the *Aging Facility Criticality Safety Calculations* document for PWR (BSC 2004a, Section 5.1.5.2) and BWR (BSC 2004a, Section 5.1.5.4) fuel.

DOE Fuel

Table 5.1-3 displays the relevant material properties for DOE non-fuel materials used in the MCNP models. Table 5.1-4 presents the isotopic content of the fuel materials for each DOE type fuel considered in this calculation.

Table 5.1-3 Material Properties for DOE Non-Fuel Materials

Material	Density (g/cm^3)	Weight Percent (wt %)	Reference/ Remark
H_2O (throughout model)	1.0 [a]	H - 0.6666667 [b] O - 0.3333333 [b]	
Magnuson Concrete	2.147	O:49.94 Ca:22.63 C:10.53 Mg:9.42 Si:4.21 K:0.9445 Al:0.7859 Fe:0.5595 Ti:0.148 Na:0.1411 H:0.3319 S:0.2483 Cl:0.0523 Mn 0.0512	NRC 1997, Volume 3, p. M8.2.4
Type 304L Stainless Steel	7.94	Fe:68.045 Cr:19.0 Ni:10.0 Mn:2.0 Si:0.75 N:0.1 P:0.045 S:0.03 C:0.03	ASTM A 276-91a 1991, p. 2 ASTM G1-90 1999, Table X1
Type 316L Stainless Steel	7.98	Fe:65.295 Cr:17.0 Ni:12.0 Mn:2.0 Mo:2.5 Si:1.0 N:0.1 P:0.045 S:0.03 C:0.03	ASTM A 276-91a 1991, p. 2 ASTM G1-90 1999, Table X1
Type 516 Carbon Steel	7.85	Fe:98.33 Mn:1.025 Si:0.275 P:0.035 S:0.035 C:0.3	ASME 2001, Sec IIA, SA-516/SA-516M & Sec IIA, SA-20/SA-20M, item 14

[a] The moderator density was varied between 0.0 – 1.0 g/cm^3 to study moderator density variations in Section 6
[b] Values given in atom fraction and not wt %

Table 5.1-4 Material Properties for Each DOE Fuel Type

DOE Fuel Type	Density (g/cm^3)	Weight Percent (wt%)	Neutron Absorber (kg)
Enrico Fermi	17.424	U-235:22.96 U-238:66.41 Mo:10.63	1.2-3.0 [a]
FFTF	10.02	O:11.63 U-235:0.13 U-238:62.37 Pu-239:22.54 Pu-240:3.01 Pu-241:0.26 Pu-242:0.06	0.4-19.26 [b]
Fort St. Vrain	1.991	Th-232:25.69 C:64.81 U-235:3.54 Si:5.96	------
Melt and Dilute	3.00	U-235:3.64 U-238:14.56 Al:77.97 Gd:0.50 H:0.37 O:2.96	0.0009-4.73 [c]
N Reactor	18.39	U-235:1.25 U-238:98.75	------
Shippingport LWBR	9.71	O:12.12 U-233:4.57 U-234:0.06 U-238:0.02 Th-232:83.23	------
Shippingport PWR – zone 1	6.36	U-235:45.04 U-238:3.29 Ca:3.72 Zr:29.54 O:18.41	------
Shippingport PWR – zone 2	6.36	U-235:32.98 U-238:2.41 Ca:4.15 Zr:39.98 O:20.48	------
Shippingport PWR – zone 3	6.36	U-235:21.74 U-238:1.59 Ca:4.57 Zr:49.67 O:22.43	------
TMI-2	10.42	U-235:2.61 U-238:85.53 O:11.86	------
TRIGA	6.58	U-235:5.94 U-238:2.56 Zr:89.91 H:1.59	------

SOURCE: BSC 2004b, Table 5-3. Also, see BSC 2004b, Section 5.1.2 for fuel description.
[a] Neutron absorber (Gd) contents in canister were varied. 1 vol% corresponds to 3 kg (CRWMS M&O 2000a, p.12)
[b] Neutron absorber (Gd) contents in canister were varied. 5 wt% corresponds to 19.26 kg, which is the maximum amount of gadolinium (CRWMS M&O 1999a, p.21)
[c] Neutron absorber (Gd) contents in ingots were varied. 0.5 wt% corresponds to 4.73 kg (BSC 2001c, p.3)

Naval Fuel Canisters

Per Assumption 3.2, no evaluations will be performed of naval fuel canisters since it is the responsibility of the U. S. Department of the Navy.

5.2 CRITICALITY CALCULATIONS

The process and methodology for criticality safety analysis given in the *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7) were implemented in these calculations. The sections below feature calculations performed in this document. As mentioned earlier, results from previously performed applicable criticality evaluations are utilized in Section 6 and are not included in sections below.

5.2.1 DOE Fuel Moderator Density Variations

Moderator density, which could vary from dry to fully moderated conditions under accident conditions, was varied in the MCNP model over the range of 0.0 to 1.0 g/cm^3. The calculations were performed for all DOE fuels considered in this document. They were each modeled as single canisters with their outsides surrounded by concrete.

5.2.2 DOE Fuel Neutron Poison Variations

Variations in neutron poison loading were studied to cover any errors of its integration in the basket structure or fuel mixture. The DOE fuels that have neutron poisons as part of their interior basket structure are Enrico Fermi and FFTF. Melt & Dilute fuel has integrated neutron poison in its fuel. Each of these DOE fuels were modeled as single canisters with their outsides surrounded by concrete. The mass of the neutron absorbers was varied in accordance with the range shown in Table 5.1-4.

5.2.3 Category 1 and 2 Event Sequences

At the present time, no Category 1 and 2 events have been identified for the Fuel Handling Facility. The *Internal Hazards Analysis for License Application* document (BSC 2004c) identifies potential events that could lead to a criticality accident. While these potential events have not yet been categorized into Category 1 and 2 or beyond Category 2 events, all of the identified potential events will be considered in this document for conservatism.

Table 5.2-1 describes the potential criticality events for the FHF entrance vestibule and the applicable criticality safety evaluation performed for each event. Table 5.2-2 describes the potential criticality events for the FHF preparation room and the applicable criticality safety evaluation performed for each event. Table 5.2-3 refers to potential criticality events associated with CSNF assembly transfer and the FHF main transfer room, the fuel transfer bay, and the fuel transfer room. Table 5.2-4 refers to potential criticality events for canister transfer in the FHF main transfer room. Table 5.2-5 describes the potential criticality events for FHF WP closure. Table 5.2-6 describes the potential criticality events for WP loadout in the FHF main transfer room, preparation room, and entrance vestibule. Table 5.2-7 refers to potential criticality events for loaded MSC removal operations in the FHF main transfer room, preparation room, and entrance vestibule. The supporting calculations for the potential criticality events are provided in the subsections.

Table 5.2-1 Potential Criticality Events for the FHF Entrance Vestibule

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with a railcar (holding a loaded cask) derailment or collision followed by a load tipover or fall and rearrangement of the cask internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for this event.
Criticality associated with an overturning or collision involving an LWT or an OWT holding a loaded cask and rearrangement of cask internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for this event.
Criticality associated with a drop or slapdown of a cask and a rearrangement of the container internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for this event.
Criticality associated with a drop or slapdown of an MSC and a rearrangement of the container internals.	Per Assumption 3.3, the MSC is similar in design to a NRC-certified cask. There is no effect on the criticality control features of the system as a result of this event shown by the cask handling accident evaluation in BSC 2004a, Table 5.2-1. Furthermore, there is no moderator intrusion to make the configuration more reactive.
Criticality associated with a gantry crane holding an MSC collision followed by a load drop or tipover and a rearrangement of the MSC internals.	Per Assumption 3.3, the MSC is similar in design to a NRC-certified cask. There is no effect on the criticality control features of the system as a result of this event shown by the cask tip-over evaluation in BSC 2004a, Table 5.2-1. Furthermore, there is no moderator intrusion to make the configuration more reactive.

[a] BSC 2004c, Table 23b

Table 5.2-2 Potential Criticality Events for the FHF Preparation Room

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with a loaded cask or loaded MSC collision or trolley derailment followed by a load tipover or fall and a rearrangement of the cask internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for this event. Also, there is no effect on the criticality control features of the system as a result of this event shown by the tip-over evaluation in BSC 2004a, Table 5.2-1. Furthermore, there is no moderator intrusion to make the configuration more reactive.

[a] BSC 2004c, Table 24b

Table 5.2-3 Potential Criticality Events for the FHF Transfer Rooms [b]

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with a drop of an SNF assembly from the spent fuel transfer machine into a cask, MSC, or WP and a rearrangement of the cask, MSC, or WP internals.	Since the transportation cask, for the SNF assembly drop into a transportation cask scenario, is NRC-certified to be 10 CFR 71 compliant, the criticality evaluation performed for the cask certification is adequate to cover this event. Per Assumption 3.3, the MSC is similar in design to a NRC certified cask, and will also be in compliance with 10 CFR 71. The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7) for the potential event involving the WP.
Criticality associated with a drop of an SNF assembly from the spent fuel transfer machine and a rearrangement of the fuel rods that comprise the assembly due to impact.	The drop could cause reconfiguration of the CSNF assembly. Section 5.2.3.1 of BSC 2004f evaluates the k_{eff} of a reconfigured, fully flooded, CSNF and it remains safely below 0.9.
Criticality associated with the drop of heavy equipment onto a loaded, open cask, MSC, or WP and a rearrangement of the container internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for the event with the transportation cask. Per Assumption 3.3, the MSC is designed to the same standards as the cask and consequently regulatory compliance provides assurance of criticality safety. The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7) for the potential event involving the WP.
Criticality associated with a misload of a WP or and MSC.	Fuel assembly misloading is not an issue for “out-of-package” criticality, as the criticality evaluations (BSC 2004a & BSC 2004f) are based on 5% maximum enrichment and no burnup credit is taken.
Docking ring leaking water into an unsealed loaded cask or MSC leads to a criticality.	Criticality evaluations for fully flooded conditions of a MSC (BSC 2004a, Section 6.3) and 5% maximum enrichment with no burnup credit taken shows that there is no criticality concern. Further, moderator intrusion studies shows (BSC 2004f, Section 6.1) that fully flooded conditions are bounding.

[a] BSC 2004c, Table 26b

[b] The events are associated with CSNF assembly transfer and applies to all FHF transfer rooms (i.e., main transfer room, fuel transfer bay, and fuel transfer room)

Table 5.2-4 Potential Criticality Events for the FHF Main Transfer Room [b]

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with a drop of a loaded cask or MSC from the main transfer room overhead crane and a rearrangement of the cask or MSC internals.	Per Assumption 3.3, the MSC is designed to the same standards as the cask and consequently regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety.
Criticality associated with a drop of a DPC, a DOE SNF canister, a naval SNF canister, or a DOE HLW canister and a rearrangement of canister internals.	Per Assumption 3.2 it is the responsibility of the U.S Department of the Navy to ensure criticality safety for the naval SNF canister. Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for the event with the DPC. Section 5.2.3.1of this document demonstrates that a rearrangement of fuel for the DOE canisters does not increase the reactivity. In addition, Section 5.2.3.2 shows that losing the skirt, or impact limiter, of the canister due to a drop also not increase the reactivity.
Criticality associated with the drop of heavy equipment onto a loaded, open cask, MSC, or WP and a rearrangement of the container internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for the event with the transportation cask. Per Assumption 3.3, the MSC is designed to the same standards as the cask and consequently regulatory compliance provides assurance of criticality safety. The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7) for the potential event involving the WP.
Criticality associated with a misload of a WP or an MSC.	Fuel assembly misloading is not an issue for "out-of-package" criticality (BSC 2004e, p. 1), as the criticality evaluations (BSC 2004a & BSC 2004f) are based on 5% maximum enrichment and no burnup credit is taken.

[a] BSC 2004c, Table 27b
[b] The potential events are associated with canister transfer

Table 5.2-5 Potential Criticality Events for FHF Waste Package Closure

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with a trolley holding a sealed or unsealed WP derailment followed by a load tipover or fall and rearrangement of the container internals.	The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7).
Criticality associated with a drop of a loaded, unsealed WP from the main transfer room overhead crane and a rearrangement of the container internals.	The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7).
Criticality associated with the drop of heavy equipment onto a loaded, unsealed WP and a rearrangement of the container internals.	The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7).

[a] BSC 2004c, Table 28b

Table 5.2-6 Potential Criticality Events for Waste Package Loadout

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with a trolley holding a sealed WP derailment followed by a load tipover or fall and rearrangement of the WP internals.	The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7).
Criticality associated with a drop or collision of a sealed WP and a rearrangement of the container internals.	The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7).
Criticality associated with a slapdown of a sealed WP and a rearrangement of the container internals.	The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7).

[a] BSC 2004c, Table 29b
[b] The events applies to the FHF main transfer room, preparation room, and entrance vestibule

Table 5.2-7 Potential Criticality Events for Loaded MSC Removal Operations [b]

Potential Event [a]	Criticality Safety Evaluation
Criticality associated with an MSC trolley collision or trolley derailment followed by a load tipover or fall and a rearrangement of the MSC internals.	Per Assumption 3.3, the MSC is similar in design to a NRC-certified cask. There is no effect on the criticality control features of the system as a result of this event shown by the cask tip-over evaluation in BSC 2004a, Table 5.2-1. Furthermore, there is no moderator intrusion to make the configuration more reactive.
Criticality associated with a drop or slapdown of a loaded MSC from an overhead crane and a rearrangement of cask internals.	Per Assumption 3.2, the MSC is similar in design to a NRC-certified cask. There is no effect on the criticality control features of the system as a result of this event shown by the cask handling accident evaluation in BSC 2004a, Table 5.2-1. Furthermore, there is no moderator intrusion to make the configuration more reactive.
Criticality associated with the drop of heavy equipment onto an unsealed MSC and a rearrangement of the container internals.	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for the event with the transportation cask. Per Assumption 3.3, the MSC is designed to the same standards as the cask and consequently regulatory compliance provides assurance of criticality safety. The WP is designed to withstand credible hazards without significant rearrangement of the fuel (Minwalla 2003, Section 4.9.2.2.7) for the potential event involving the WP.

[a] BSC 2004c, Table 30b
[b] The events applies to the FHF main transfer room, preparation room, and entrance vestibule

5.2.3.1 Rearrangement of DOE Canister Internals

Potential Category 1 and 2 event sequences includes drops of DOE canisters causing a rearrangement of canister internals (see Table 5.2-4). For this purpose fuel pin pitch variations (i.e., increased and deceased pin pitch) were modeled for the applicable DOE canisters in MCNP to simulate this potential event. The DOE canisters were modeled as single canisters with flooded inside conditions and concrete reflection outside, representing the most reactive configuration. Table 6-1 of BSC 2004b demonstrates that for nearly all DOE fuel types, k_{eff}

increases when the outside is surrounded by concrete instead of water. The reason for this is that concrete is a better reflector than water in the environment that is being modeled. Table 5.2-8 lists the k_{eff} as a function of pin pitch for each applicable DOE fuel types in their respective canisters. It can be seen that varying the pin pitch does not cause a criticality concern. Further, the pin pitches that produce the maximum k_{eff} are used for criticality evaluations under normal conditions (BSC 2004b, Section 6.1). The results presented in Table 5.2-8 are also illustrated in Figure 5.2-1, which is generated in *DOE_fuel.xls*. Note that the N Reactor fuel denoted "A" in the figure refers to Mark 1A fuel and "B" refers to Mark IV fuel.

Table 5.2-8 DOE Fuel Pin Pitch Variations

Pitch (cm)	Keff	St. Dev	MCNP files [a]	Pitch (cm)	Keff	St. Dev	MCNP files [a]
	Enrico Fermi				FFTF		
0.01	0.81681	0.00077	efwds01.out	0.6	0.83516	0.00094	ffwds06.out
0.02	0.83621	0.00076	efwds02.out	0.8	0.82977	0.00097	ffwds08.out
0.03	0.85551	0.00076	efwds03.out	1.0	0.8512	0.00095	ffwds10.out
0.04	0.87361	0.00074	efwds04.out	1.2	0.86941	0.00091	ffwds12.out
0.05	0.88478	0.00077	efwds05.out	1.4	0.87562	0.00102	ffwds14.out
0.06	0.88616	0.00078	efwds06.out	1.5	0.87592	0.00076	ffwds15.out
0.07	0.88564	0.00077	efwds07.out	1.6	0.87644	0.00103	ffwds16.out
0.08	0.88028	0.00079	efwds08.out	1.7	0.87629	0.0008	ffwds17.out
0.09	0.87757	0.00077	efwds09.out	1.8	0.87231	0.00094	ffwds18.out
0.1	0.87865	0.00085	efwds10.out	2.0	0.86517	0.00106	ffwds20.out
0.11	0.88474	0.00081	efwds11.out		TRIGA		
0.12	0.88278	0.0008	efwds12.out	6.03 [b]	0.83818	0.00112	trwds60.out
0.13	0.87078	0.0008	efwds13.out	6.5	0.79473	0.00110	trwds65.out
0.14	0.85951	0.00083	efwds14.out	7.0	0.74618	0.00980	trwds70.out
0.15	0.85611	0.00075	efwds15.out	7.5	0.69476	0.00100	trwds75.out
	N Reactor Mark 1A Fuel				N Reactor Mark IV Fuel		
6.2	0.72258	0.00063	nrwds62a.out	6.2	0.71364	0.00058	nrwds62b.out
6.6	0.78310	0.00064	nrwds66a.out	6.6	0.77336	0.00060	nrwds66b.out
7.0	0.82639	0.00063	nrwds70a.out	7.0	0.81915	0.00058	nrwds70b.out
7.4	0.84729	0.00064	nrwds74a.out	7.4	0.84396	0.00059	nrwds74b.out
7.8	0.84929	0.00067	nrwds78a.out	7.8	0.85334	0.00056	nrwds78b.out
8.2	0.83357	0.00067	nrwds82a.out	8.2	0.84373	0.00058	nrwds82b.out
8.6	0.80056	0.00065	nrwds86a.out	8.6	0.82394	0.00057	nrwds86b.out
9.0	0.76490	0.00064	nrwds90a.out	9.0	0.80148	0.00055	nrwds90b.out
	TMI-2 Type D Canister				TMI-2 Type K Canister		
1.1	0.61646	0.0009	tmwds11d.out	1.1	0.41605	0.0007	tmwds11k.out
1.3	0.77349	0.00097	tmwds13d.out	1.3	0.57354	0.00087	tmwds13k.out
1.5	0.85634	0.00097	tmwds15d.out	1.5	0.70058	0.00087	tmwds15k.out
1.7	0.85296	0.00091	tmwds17d.out	1.7	0.77586	0.00099	tmwds17k.out
1.9	0.83542	0.00097	tmwds19d.out	1.9	0.80811	0.00089	tmwds19k.out
2.1	0.78778	0.00096	tmwds21d.out	2.1	0.78911	0.00097	tmwds21k.out
2.3	0.74795	0.00086	tmwds23d.out	2.3	0.73864	0.00088	tmwds23k.out
2.5	0.69909	0.00084	tmwds25d.out	2.5	0.68338	0.0009	tmwds25k.out

[a] The input files to each run have the same name as the corresponding output file but without the .out extension (e.g., the input file matching output file tmwds11d.out is tmwds11d).
[b] Smallest possible physical spacing.

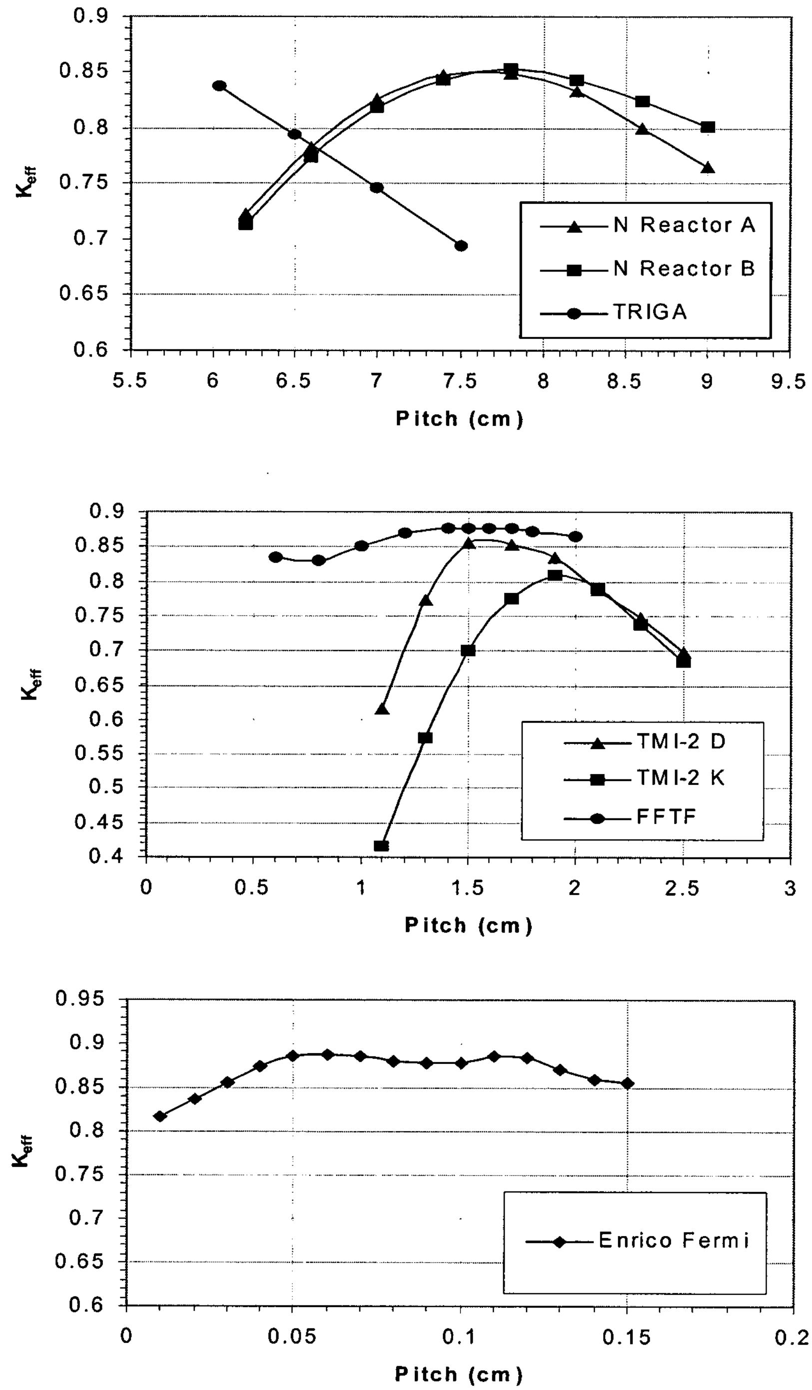

Figure 5.2-1 DOE Fuel Pin Pitches versus k_{eff}

5.2.3.2 DOE Fuel Drop or Slapdown

In the event of a drop or slapdown, as discussed earlier in Section 5.2.3.1, the lower skirts of the canister might get damaged causing the interior fuel basket to sit directly on the bottom of the canister. A calculation was performed with a single flooded TRIGA fuel canister featuring a neglected lower skirt (the area below the skirt was exchanged from water to concrete). Table 5.2-9 shows the results and it can be seen by comparing this calculation to the calculation of an intact TRIGA fuel canister that a lost or damaged lower skirt does not impact k_{eff}.

Table 5.2-9 TRIGA Fuel Canister With and Without Skirt

MCNP Model Description	K_{eff}	Standard Deviation	MCNP input & output files
TRIGA			
Intact Lower Skirt	0.83804	0.00115	trwds60, trwds60.out
Damaged Lower Skirt	0.83804	0.00115	trwds60s, trwds60s.out

6. RESULTS AND CONCLUSIONS

This section presents the results of the criticality calculations and makes recommendations for additional criticality safety design features as appropriate. The outputs presented in this document are all reasonable compared to the inputs and the results are suitable for the intended use. The uncertainties are taken into account by consistently using a conservative approach, which is the result of the methods and assumptions described in Sections 2 and 3, respectively.

6.1 CASK AND MSC CRITIALITY EVALUATION

Criticality evaluations performed in the *Aging Facility Criticality Safety Calculations* document (BSC 2004a) feature the Holtec HI-STORM 100 cask system as a representative cask. Per Assumption 3.3, the MSC is similar in design to this NRC-licensed cask. The results presented in Section 6 of BSC 2004a shows that there are no criticality concerns associated with the HI-STORM cask or the MSC. The results lead to the following main conclusions (BSC 2004a, Section 6.5):

- Both the PWR and BWR results consistently demonstrate that the conditions outside the HI-STORM overpack (e.g., spacing, moderation, and reflection) have no discernable impact on the reactivity of the cask. This indicates that the casks are neutronically isolated and consequently the cask orientation (e.g., vertical versus horizontal) will not matter.

- Reactivity of the loaded casks decreases with reduction in moderator density.

- Maximum reactivity is reached when the casks are fully flooded with water at full density (1.0 g/cm^3).

The scenarios considered in BSC 2004a covers the conditions featured in the FHF. For this purpose, no additional criticality calculations were performed for a cask or MSC in the FHF.

6.2 DOE FUEL CANISTER CRITICALITY EVALUATIONS

Criticality evaluations were performed for DOE fuel in the *Canister Handling Facility Criticality Safety Calculations* document (BSC 2004b) were the results lead to the following conclusions applicable to FHF operations (BSC 2004b, Section 6.3):

- Criticality is not a concern for any single DOE fuel canister under both normal and flooded conditions.

- Criticality is not a concern for damaged fuel resulting from a catastrophic drop of a single fuel canister for any DOE type. This is based on the assumptions that interior basket, if present, remains intact following a canister drop (Assumption 3.5) and that the damaged fuel inside the canister is dry (Assumption 3.6).

To verify that fully flooded conditions are the most reactive, additional calculations were performed were the fuel moderator density was varied. The results from the calculations are presented below in Section 6.2.1. Further, the Melt & Dilute has integrated neutron poison (Gd) in its fuel while the Enrico Fermi and FFTF fuels have neutron absorbers in their interior baskets. Variations in neutron poison loading were also studied to cover any errors of its integration in the basket structure or fuel mixture. The results from the calculations are presented below in Section 6.2.2.

6.2.1 DOE Fuel Moderator Density Variations

Moderator density, which could vary from dry to fully moderated conditions under accident conditions, has been varied over the range of 0.0 to 1.0 g/cm^3 for all DOE fuel types considered in this document. Table 6.2-1 displays k_{eff} as a function of moderator density for the various DOE fuel. It can be seen that the reactivity of the DOE fuel decreases with reduction in moderator density. Consequently, it can be concluded that fully flooded conditions are the most reactive. Figure 6.2-1, generated in *DOE_fuel.xls*, illustrates the results presented in Table 6.2-1.

Table 6.2-1 K_{eff} as a Function of Moderator Density Variations

Density (g/cc)	K_{eff}	St. Dev	MCNP files [a]	K_{eff}	St. Dev	MCNP files [a]
	Enrico Fermi			FFTF		
1.0	0.91063	0.00078	efwds06x.out	0.92084	0.00074	ffwds15y.out
0.98	0.90684	0.00078	efwds698.out	0.91541	0.00078	ffwds98.out
0.95	0.90194	0.00079	efwds695.out	0.90891	0.00082	ffwds95.out
0.90	0.88938	0.00079	efwds069.out	0.89624	0.00079	ffwds159.out
0.80	0.86651	0.00075	efwds068.out	0.87326	0.00075	ffwds158.out
0.50	0.78822	0.00076	efwds650.out	0.79055	0.00073	ffwds50.out
0.0	0.49484	0.00041	efdds01x.out	0.56406	0.00054	ffwds06y.out
	Fort St. Vrain			Melt & Dilute		
1.0	0.84467	0.00093	fswds00.out	0.83129	0.00268	mdwds00z.out
0.98	0.84357	0.00092	fswds098.out	0.82582	0.00253	mdwds098.out
0.95	0.84218	0.00095	fswds095.out	0.82558	0.00243	mdwds095.out
0.90	0.83603	0.00100	fswds009.out	0.81020	0.00261	mdwds009.out
0.80	0.82287	0.00101	fswds008.out	0.80444	0.00268	mdwds008.out
0.50	0.76258	0.00104	fswds050.out	0.75754	0.0022	mdwds050.out
0.0	0.51526	0.00099	fsdds00.out	0.65896	0.00227	mddds00z.out

Table 6.2-1(cont.) K_{eff} as a Function of Moderator Density Variations

Density (g/cc)	K_{eff}	St. Dev	MCNP files [a]	K_{eff}	St. Dev	MCNP files [a]
	N Reactor "A"			N Reactor "B"		
1.0	0.84929	0.00067	nrwds78a.out	0.85334	0.00056	nrwds78b.out
0.98	0.84736	0.00062	nrwd98a.out	0.84978	0.00058	nrwd98b.out
0.95	0.84189	0.00066	nrwd95a.out	0.84547	0.00056	nrwd95b.out
0.90	0.83229	0.00064	nrwd789a.out	0.83613	0.00059	nrwd789b.out
0.80	0.81078	0.00060	nrwd788a.out	0.81506	0.00060	nrwd788b.out
0.50	0.70305	0.00056	nrwd50a.out	0.70516	0.00058	nrwd50b.out
0.0	0.34216	0.00038	nrdds62a.out	0.34489	0.00034	nrdds62b.out
	Shippingport LWBR			Shippingport PWR		
1.0	0.87003	0.00101	slwds94.out	0.87972	0.00100	spwds00.out
0.98	0.85963	0.00112	slwds98.out	0.87777	0.00111	spwds98.out
0.95	0.84579	0.00110	slwds95.out	0.86895	0.00102	spwds95.out
0.90	0.82493	0.00105	slwds949.out	0.86277	0.00103	spwds009.out
0.80	0.77979	0.00104	slwds948.out	0.84687	0.00107	spwds008.out
0.50	0.61967	0.00096	slwds50.out	0.78996	0.00105	spwds50.out
0.0	0.22755	0.00045	sldds78.out	0.23594	0.00044	spdds00.out
	TMI-2 "D"			TMI-2 "K"		
1.0	0.85634	0.00097	tmwds15d.out	0.80811	0.00089	tmwds19k.out
0.98	0.84905	0.00091	tmwd98d.out	0.80082	0.00098	tmwd98k.out
0.95	0.83840	0.00094	tmwd95d.out	0.79464	0.00089	tmwd95k.out
0.90	0.81916	0.00106	tmwd159d.out	0.78005	0.00093	tmwd199k.out
0.80	0.77684	0.00100	tmwd158d.out	0.74678	0.00097	tmwd198k.out
0.50	0.60730	0.00094	tmwd50d.out	0.60890	0.00094	tmwd50k.out
0.0	0.21237	0.00035	tmdds11d.out	0.18203	0.00032	tmdds15k.out
	TRIGA					
1.0	0.83818	0.00112	trwds60.out			
0.98	0.83600	0.00105	trwds98.out			
0.95	0.83750	0.00106	trwds95.out			
0.90	0.83361	0.00104	trwds609.out			
0.80	0.82977	0.00112	trwds608.out			
0.50	0.80371	0.00104	trwds50.out			
0.0	0.70702	0.00093	trdds60.out			

[a] The input files to each run have the same name as the corresponding output file but without the .out extension (e.g., the input file matching output file trwds60.out is trwds60).

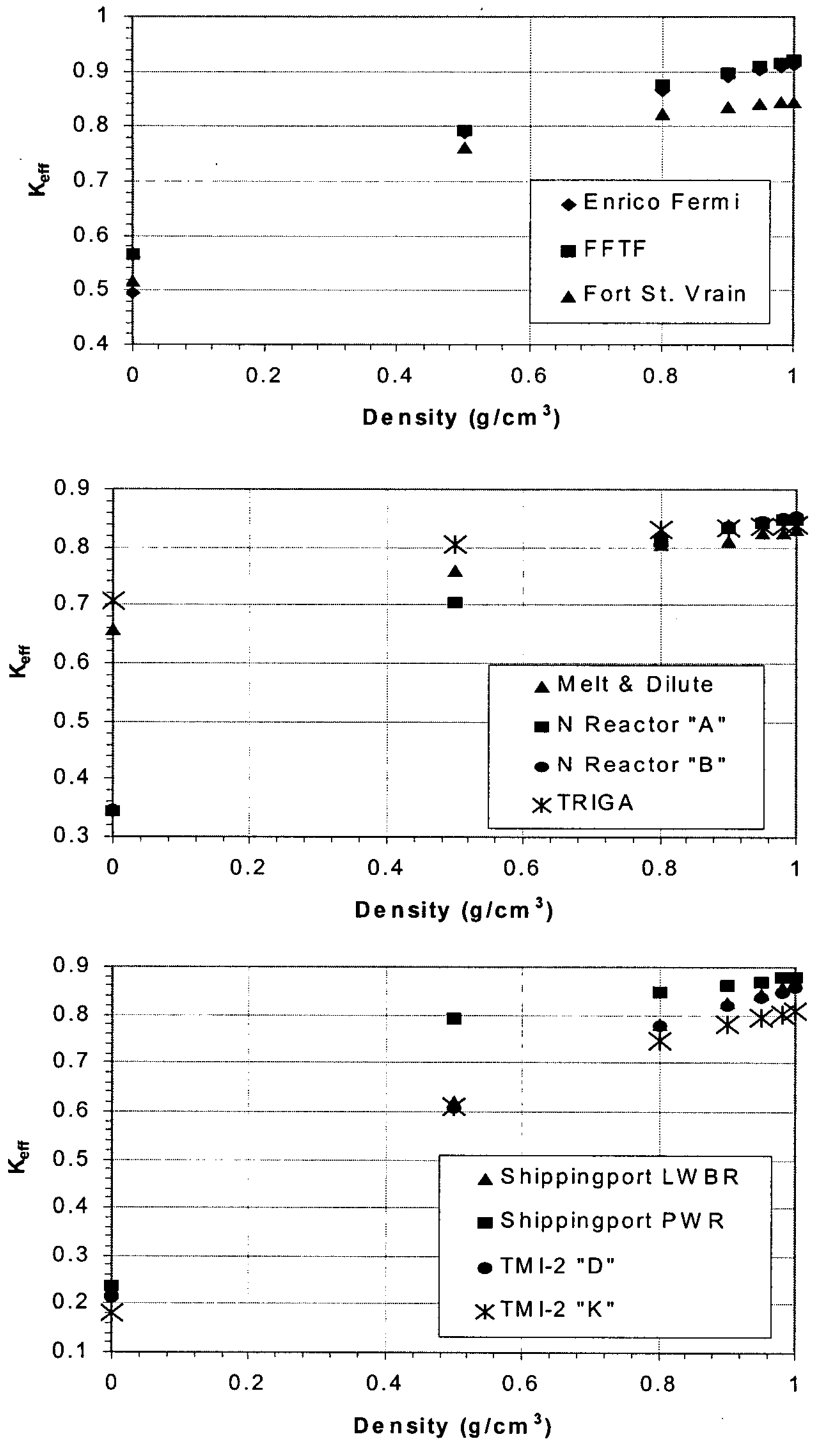

Figure 6.2-1 DOE Fuel Moderator Density Variations versus k_{eff}

6.2.2 DOE Fuel Neutron Poison Variations

Variations in neutron poison loading were studied, to cover any errors of its integration in the basket structure or fuel mixture, and Table 6.2-2 shows the variations in k_{eff} as a function of neutron poison loading. Note that the Enrico Fermi and FFTF fuels have non-integrated neutron poison (i.e., the neutron poison is fixed and a part of the basket structure) while the Melt & Dilute fuel has integrated neutron poison in its fuel. It can be seen from Table 6.2-2 that a large decrease in neutron poison will not cause k_{eff} of a single canister to exceed the upper subcritical limit.

Table 6.2-2 DOE Fuel Neutron Poison Variations

Neutron Poison (vol%)	Keff	St. Dev	MCNP files [a]	Neutron Poison (wt%)	Keff	St. Dev	MCNP files [a]
	Enrico Fermi (non-integrated neutron poison)				FFTF (non-integrated neutron poison)		
1.0 (3 kg)	0.88616	0.00078	efwds06.out	5.0 (19.26 kg)	0.87592	0.00076	ffwds15.out
0.4 (1.2 kg)	0.91063	0.00078	efwds06x.out	0.1 (0.39 kg)	0.92084	0.00074	ffwds15y.out
[a] The input files to each run have the same name as the corresponding output file but without the .out extension (e.g., the input file matching output file mdwds00.out is mdwds00).					Melt & Dilute (integrated neutron poison)		
				0.5 (4.73 kg)	0.39017	0.00124	mdwds00.out
				0.001 (0.009 kg)	0.80902	0.00256	mdwds00x.out
				0.0001 (9E-4 kg)	0.83129	0.00268	mdwds00z.out

6.3 CATEGORY 1 AND 2 EVENT SEQUENCES

Category 1 and 2 event sequences were evaluated as presented in Section 5.2.3 and were found to be within the criticality safety design limits.

6.4 CONCLUSIONS AND RECOMMENDATIONS

The FHF and its processes have been evaluated for criticality safety for normal operations, Category 1 and 2 event sequences. The results presented in this document lead to the following conclusions and recommendations:

- The criticality evaluations of the cask and MSC demonstrate that the conditions outside the overpack (e.g., spacing, moderation, reflection) have no discernable impact on the reactivity of the cask for both PWR and BWR fuel (also noted in Section 6.1).

- Single DOE canisters are subcritical under both normal and flooded conditions. Concrete surrounding the DOE canister is a better reflector than water and consequently produces a higher k_{eff}.

- The DOE fuel containing neutron poison (ie., Enrico Fermi, FFTF, and Melt & Dilute) showed that losing a significant amount of neutron poison (due to manufacturing errors etc.) still promotes subcriticality.

- Reactivity of the loaded cask, MSC or DOE canisters decreases with reduction in moderator density.

- Maximum reactivity is reached when the cask, MSC or DOE canisters are fully flooded with water at full density (1.0 g/cm^3).

- Criticality events potentially occurring in the FHF do not compromise criticality safety. It should be recognized that the potential criticality events have not yet been categorized into Category 1 and 2 event sequences.

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8. ATTACHMENTS

This calculation document includes three attachments:

ATTACHMENT I	Listing of Computer Files (7 pages)
ATTACHMENT II	One Compact Disk Containing All Files Listed in Attachment I (1 of 1) (0 pages)
ATTACHMENT III	Draft Sketches of FHF Moderator Control Areas (10 pages)

ATTACHMENT I
LISTING OF COMPUTER FILES

This attachment lists the input and output file names for the MCNP and Excel calculations. All input and output are stored on an electronic medium (compact disc) in ASCII format as part of this attachment.

Date	Time	File Size	File Name
06/07/2004	04:23p	78,848	DOE_fuel.xls
06/08/2004	12:11p	5,961	trwds60
06/08/2004	12:11p	474,856	trwds60.out
06/08/2004	12:11p	5,999	trwds60s
06/08/2004	12:11p	476,188	trwds60s.out
02/26/2004	04:54p	11,905	efwds068
02/26/2004	10:23p	654,776	efwds068.out
02/26/2004	04:54p	11,905	efwds069
02/26/2004	05:50p	654,776	efwds069.out
06/08/2004	12:09p	11,918	efwds650
06/08/2004	12:09p	654,956	efwds650.out
06/08/2004	12:09p	11,918	efwds695
06/08/2004	12:09p	654,642	efwds695.out
06/08/2004	12:09p	11,918	efwds698
06/08/2004	12:09p	654,743	efwds698.out
02/26/2004	04:55p	15,398	ffwds158
02/26/2004	11:26p	676,100	ffwds158.out
02/26/2004	04:55p	15,398	ffwds159
02/26/2004	06:51p	676,100	ffwds159.out
06/08/2004	12:10p	15,402	ffwds50
06/08/2004	12:10p	676,601	ffwds50.out
06/08/2004	12:10p	15,402	ffwds95
06/08/2004	12:10p	676,178	ffwds95.out
06/08/2004	12:10p	15,402	ffwds98
06/08/2004	12:10p	676,279	ffwds98.out
02/26/2004	04:57p	11,053	fswds008
02/26/2004	11:57p	520,020	fswds008.out
02/26/2004	04:57p	11,053	fswds009
02/26/2004	07:23p	520,020	fswds009.out
06/08/2004	12:10p	11,083	fswds050
06/08/2004	12:10p	519,950	fswds050.out
06/08/2004	12:10p	11,069	fswds095
06/08/2004	12:10p	520,066	fswds095.out
06/08/2004	12:10p	11,083	fswds098
06/08/2004	12:10p	519,950	fswds098.out
02/26/2004	04:59p	5,540	mdwds008
02/27/2004	12:00a	342,086	mdwds008.out

Date	Time	File Size	File Name
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02/26/2004	07:26p	342,086	mdwds009.out
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06/08/2004	12:10p	342,085	mdwds050.out
06/08/2004	12:10p	5,539	mdwds095
06/08/2004	12:10p	342,085	mdwds095.out
06/08/2004	12:10p	5,539	mdwds098
06/08/2004	12:10p	342,220	mdwds098.out
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06/08/2004	12:10p	413,605	nrwd50a.out
06/08/2004	12:10p	7,324	nrwd50b
06/08/2004	12:10p	414,896	nrwd50b.out
02/26/2004	05:03p	7,302	nrwd788a
02/27/2004	12:20a	415,067	nrwd788a.out
02/26/2004	05:04p	7,324	nrwd788b
02/27/2004	12:39a	413,820	nrwd788b.out
02/26/2004	05:03p	7,302	nrwd789a
02/26/2004	07:45p	415,067	nrwd789a.out
02/26/2004	05:04p	7,324	nrwd789b
02/26/2004	08:04p	414,972	nrwd789b.out
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06/08/2004	12:10p	414,439	nrwd95a.out
06/08/2004	12:10p	7,324	nrwd95b
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06/08/2004	12:10p	7,302	nrwd98a
06/08/2004	12:10p	414,475	nrwd98a.out
06/08/2004	12:10p	7,324	nrwd98b
06/08/2004	12:10p	414,896	nrwd98b.out
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06/08/2004	12:10p	491,718	slwds50.out
02/26/2004	05:05p	11,741	slwds948
02/27/2004	01:07a	491,504	slwds948.out
02/26/2004	05:05p	11,741	slwds949
02/26/2004	08:31p	491,504	slwds949.out
06/08/2004	12:10p	11,758	slwds95
06/08/2004	12:10p	491,765	slwds95.out
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06/08/2004	12:10p	491,718	slwds98.out
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02/27/2004	01:50a	796,106	spwds008.out
02/26/2004	05:05p	27,582	spwds009
02/26/2004	09:15p	796,106	spwds009.out
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06/08/2004	12:10p	796,371	spwds050.out

Date	Time	File Size	File Name
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02/26/2004	09:31p	445,916	tmwd159d.out
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02/27/2004	07:20a	467,153	tmwd198k.out
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06/08/2004	12:10p	467,524	tmwd95k.out
06/08/2004	12:10p	6,457	tmwd98d
06/08/2004	12:10p	445,969	tmwd98d.out
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02/26/2004	05:07p	5,989	trwds609
02/26/2004	09:46p	463,333	trwds609.out
06/08/2004	12:10p	5,981	trwds95
06/08/2004	12:10p	464,064	trwds95.out
06/08/2004	12:10p	5,982	trwds98
06/08/2004	12:10p	464,162	trwds98.out
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06/08/2004	01:09p	654,644	efwds01.out
06/08/2004	01:10p	11,878	efwds02
06/08/2004	01:10p	654,157	efwds02.out
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06/08/2004	01:10p	654,776	efwds06.out
06/08/2004	01:10p	11,879	efwds06x
06/08/2004	01:10p	654,776	efwds06x.out
06/08/2004	01:10p	11,878	efwds07
06/08/2004	01:10p	655,167	efwds07.out
06/08/2004	01:10p	11,878	efwds08
06/08/2004	01:10p	655,209	efwds08.out
06/08/2004	01:10p	11,878	efwds09
06/08/2004	01:10p	653,726	efwds09.out
06/08/2004	01:10p	11,877	efwds10
06/08/2004	01:10p	654,458	efwds10.out
06/08/2004	01:10p	11,878	efwds11
06/08/2004	01:10p	654,442	efwds11.out
06/08/2004	01:10p	11,878	efwds12
06/08/2004	01:10p	654,458	efwds12.out
06/08/2004	01:10p	11,878	efwds13
06/08/2004	01:10p	654,458	efwds13.out
06/08/2004	01:10p	11,878	efwds14
06/08/2004	01:10p	654,458	efwds14.out
06/08/2004	01:10p	11,878	efwds15
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06/08/2004	01:10p	509,731	ffwds06.out
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06/08/2004	01:10p	509,731	ffwds10.out
06/08/2004	01:10p	15,378	ffwds12
06/08/2004	01:10p	509,731	ffwds12.out
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06/08/2004	01:10p	509,413	ffwds14.out
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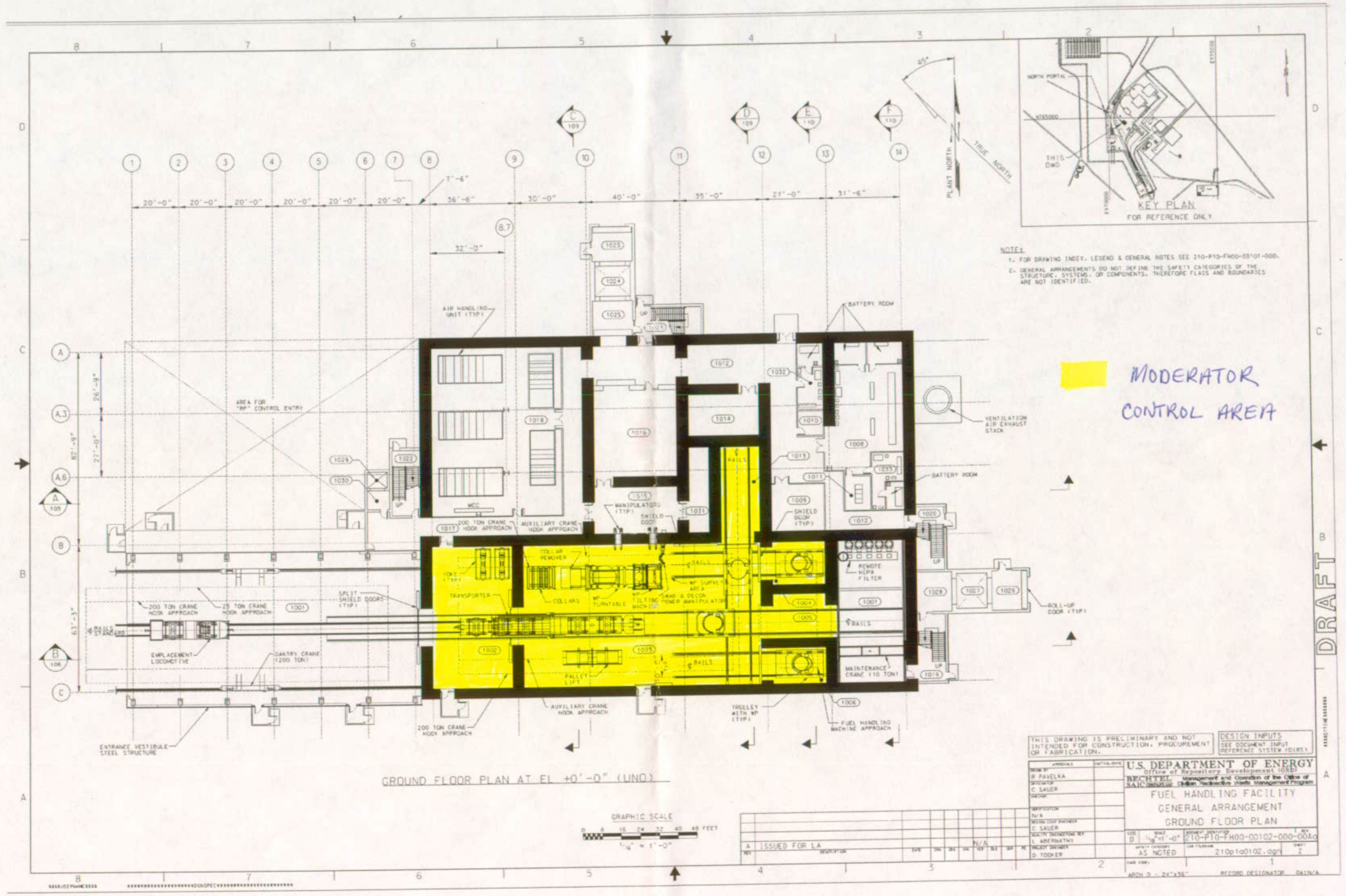
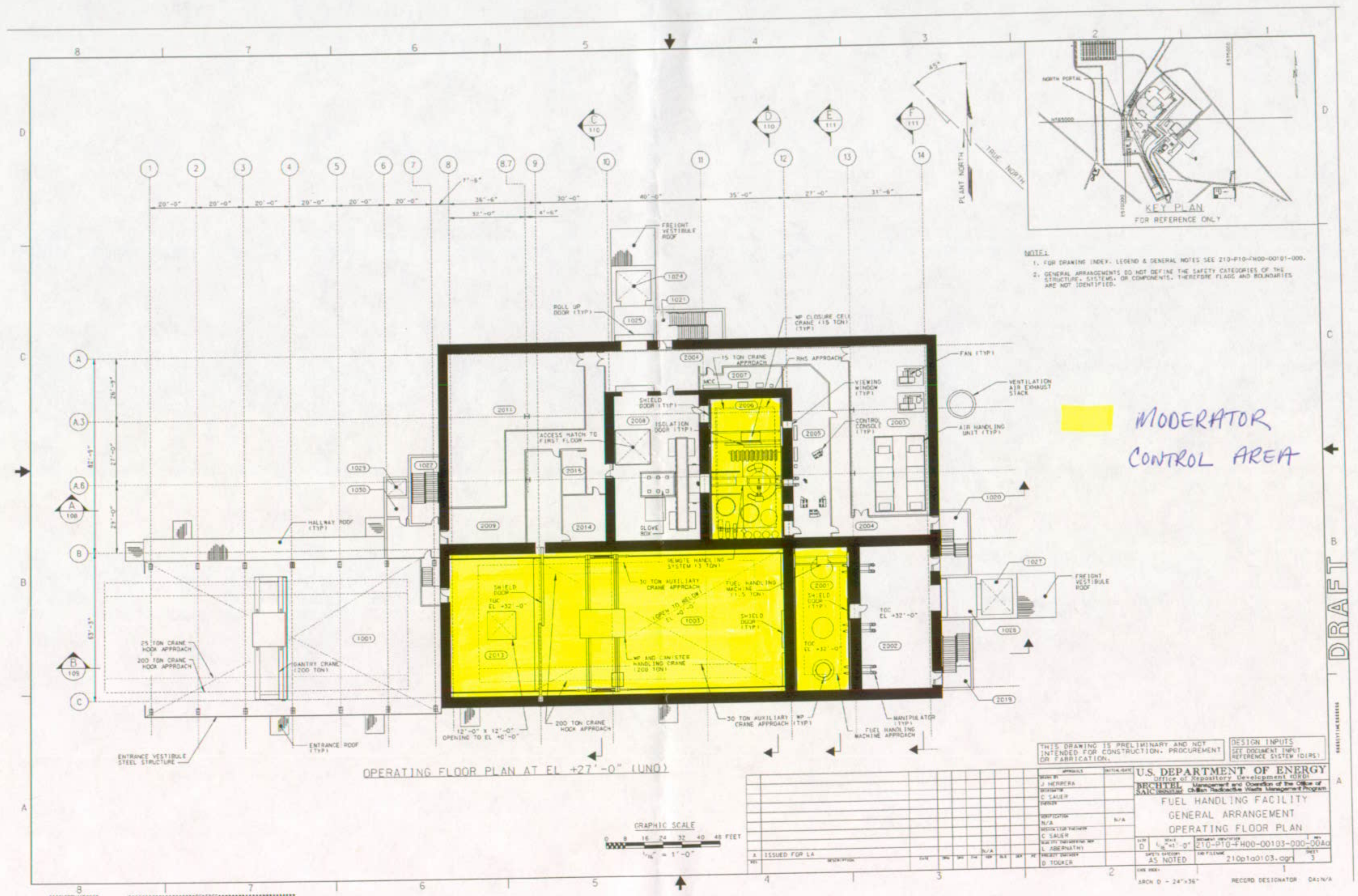

MODERATOR
CONTROL AREA
KEY PLAN
FOR REFERENCE ONLY
GROUND FLOOR PLAN AT EL +0'-0" (UNO)
GRAPHIC SCALE
DRAFT
U.S. DEPARTMENT OF ENERGY
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
GROUND FLOOR PLAN
ISSUED FOR LA


MODERATOR CONTROL AREA
OPERATING FLOOR PLAN AT EL +27'-0" (UNO)
KEY PLAN
FOR REFERENCE ONLY
DRAFT
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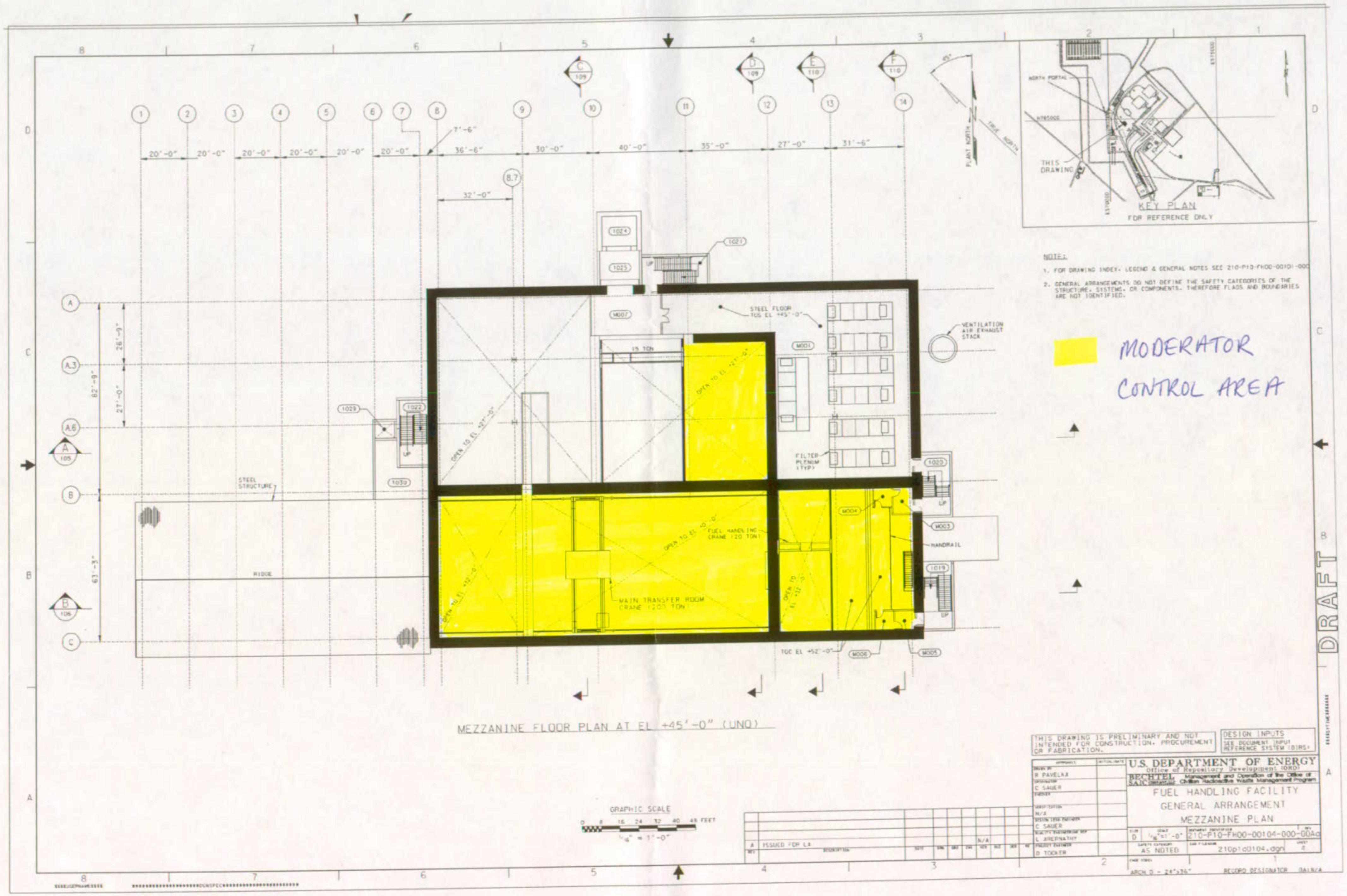
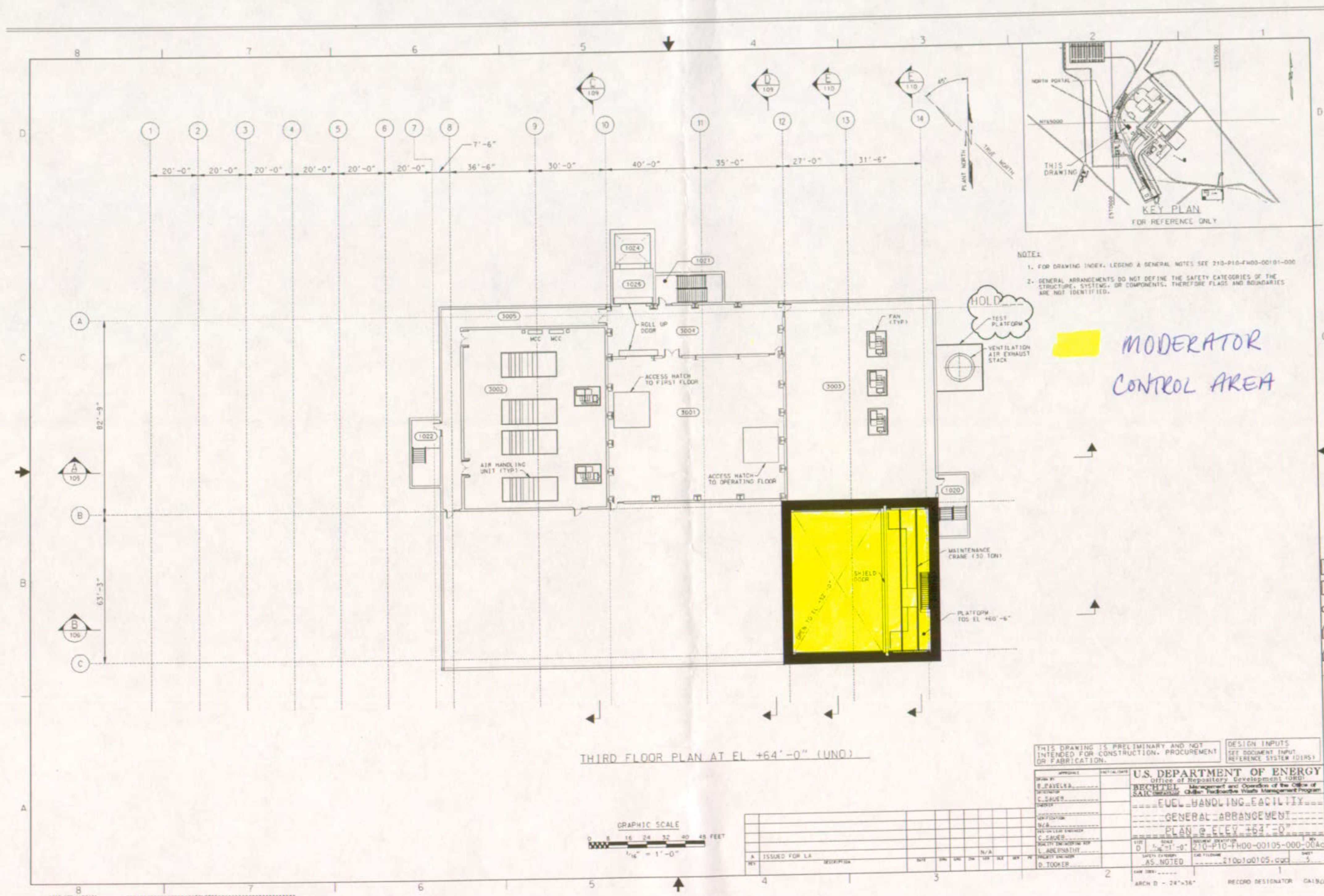
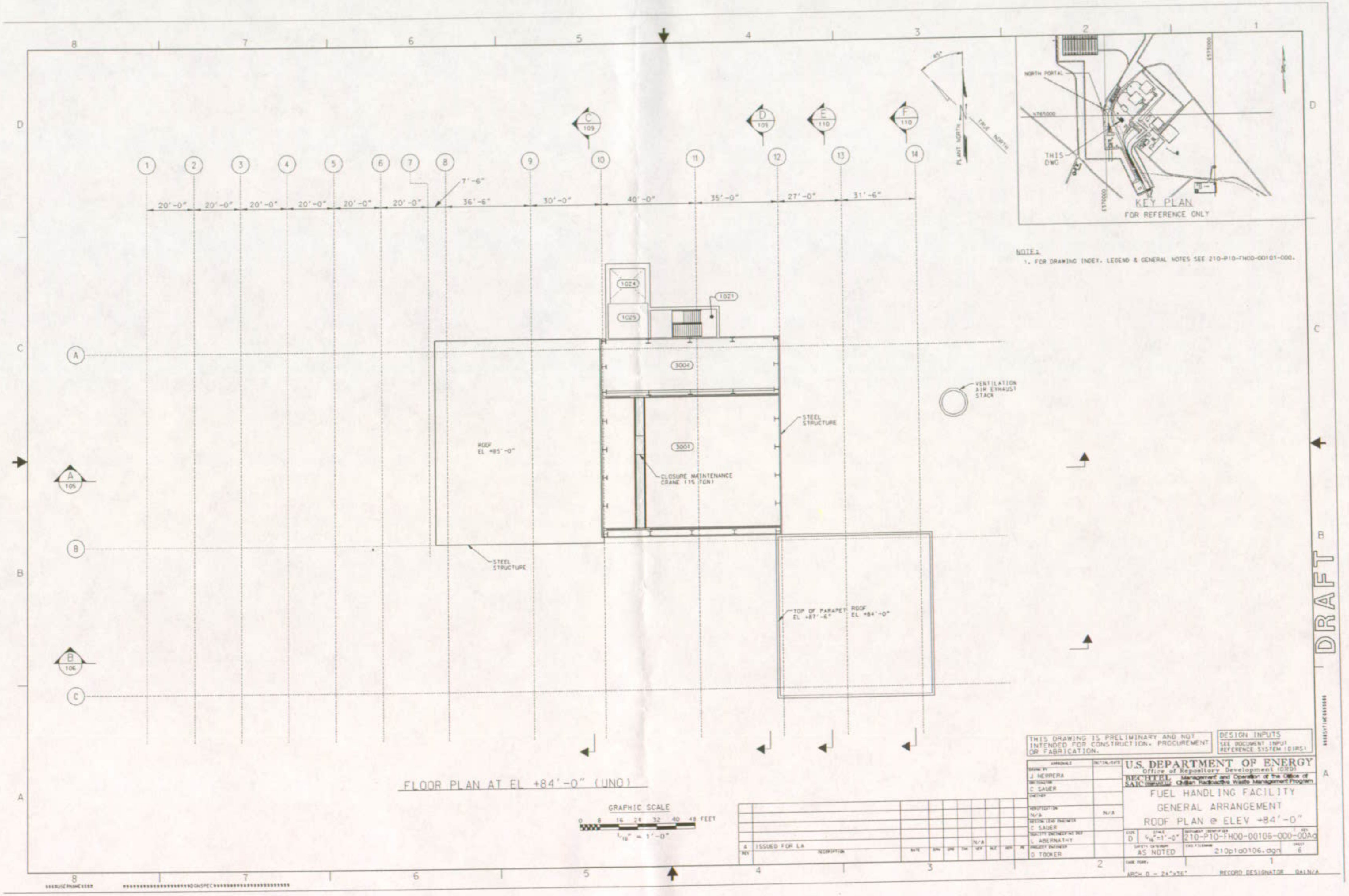
U.S. DEPARTMENT OF ENERGY
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
OPERATING FLOOR PLAN
210-P10-FH00-00103-000-00A
GRAPHIC SCALE


MODERATOR CONTROL AREA
MEZZANINE FLOOR PLAN AT EL +45'-0" (UNO)
DRAFT
KEY PLAN
FOR REFERENCE ONLY
U.S. DEPARTMENT OF ENERGY
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
MEZZANINE PLAN
210-P10-FH00-00104-000-00A
MAIN TRANSFER ROOM CRANE (20 TON)
FUEL HANDLING CRANE (20 TON)
GRAPHIC SCALE


KEY PLAN
FOR REFERENCE ONLY
NOTES:
1. FOR DRAWING INDEX, LEGEND & GENERAL NOTES SEE 210-P10-FH00-00101-000
2. GENERAL ARRANGEMENTS DO NOT DEFINE THE SAFETY CATEGORIES OF THE STRUCTURE, SYSTEMS, OR COMPONENTS. THEREFORE FLAGS AND BOUNDARIES ARE NOT IDENTIFIED.
MODERATOR CONTROL AREA
HOLD
TEST PLATFORM
VENTILATION AIR EXHAUST STACK
ROLL UP DOOR
ACCESS HATCH TO FIRST FLOOR
ACCESS HATCH TO OPERATING FLOOR
AIR HANDLING UNIT (TYP)
FAN (TYP)
MAINTENANCE CRANE (30 TON)
PLATFORM TOS EL +60'-6"
SHIELD DOOR
THIRD FLOOR PLAN AT EL +64'-0" (UNO)
GRAPHIC SCALE
DRAFT
U.S. DEPARTMENT OF ENERGY
BECHTEL SAIC
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
210-P10-FH00-00105-000-00A
AS NOTED
ISSUED FOR LA


KEY PLAN
FOR REFERENCE ONLY
NOTE:
1. FOR DRAWING INDEX, LEGEND & GENERAL NOTES SEE 210-P10-FH00-00101-000.
VENTILATION AIR EXHAUST STACK
STEEL STRUCTURE
CLOSURE MAINTENANCE CRANE (15 TON)
ROOF EL +85'-0"
TOP OF PARAPET EL +87'-6"
ROOF EL +84'-0"
FLOOR PLAN AT EL +84'-0" (UNO)
GRAPHIC SCALE
DRAFT
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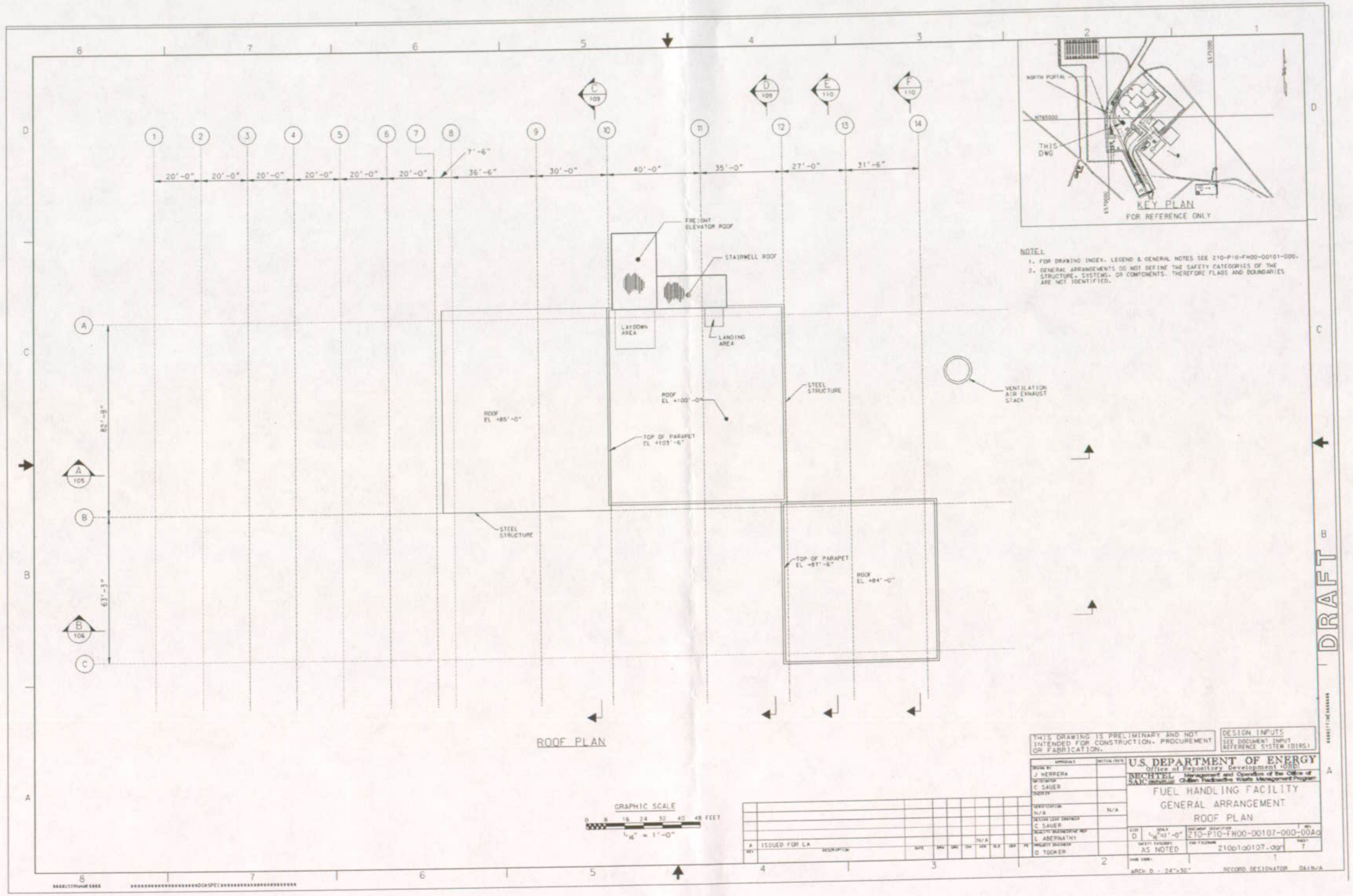
DESIGN INPUTS
U.S. DEPARTMENT OF ENERGY
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
ROOF PLAN @ ELEV +84'-0"
210-P10-FH00-00106-000-00A
AS NOTED
ISSUED FOR LA


KEY PLAN
FOR REFERENCE ONLY
THIS DWG
NORTH PORTAL
NOTE:
1. FOR DRAWING INDEX, LEGEND & GENERAL NOTES SEE 210-P10-FH00-00101-000.
2. GENERAL ARRANGEMENTS DO NOT DEFINE THE SAFETY CATEGORIES OF THE STRUCTURE, SYSTEMS, OR COMPONENTS. THEREFORE FLAGS AND BOUNDARIES ARE NOT IDENTIFIED.
20'-0"
20'-0"
20'-0"
20'-0"
20'-0"
20'-0"
7'-6"
36'-6"
30'-0"
40'-0"
35'-0"
27'-0"
31'-6"
82'-9"
63'-3"
FREIGHT ELEVATOR ROOF
STAIRWELL ROOF
LAYDOWN AREA
LANDING AREA
ROOF EL +85'-0"
ROOF EL +100'-0"
TOP OF PARAPET EL +103'-6"
STEEL STRUCTURE
VENTILATION AIR EXHAUST STACK
TOP OF PARAPET EL +87'-6"
ROOF EL +84'-0"
ROOF PLAN
GRAPHIC SCALE
1/16" = 1'-0"
DRAFT
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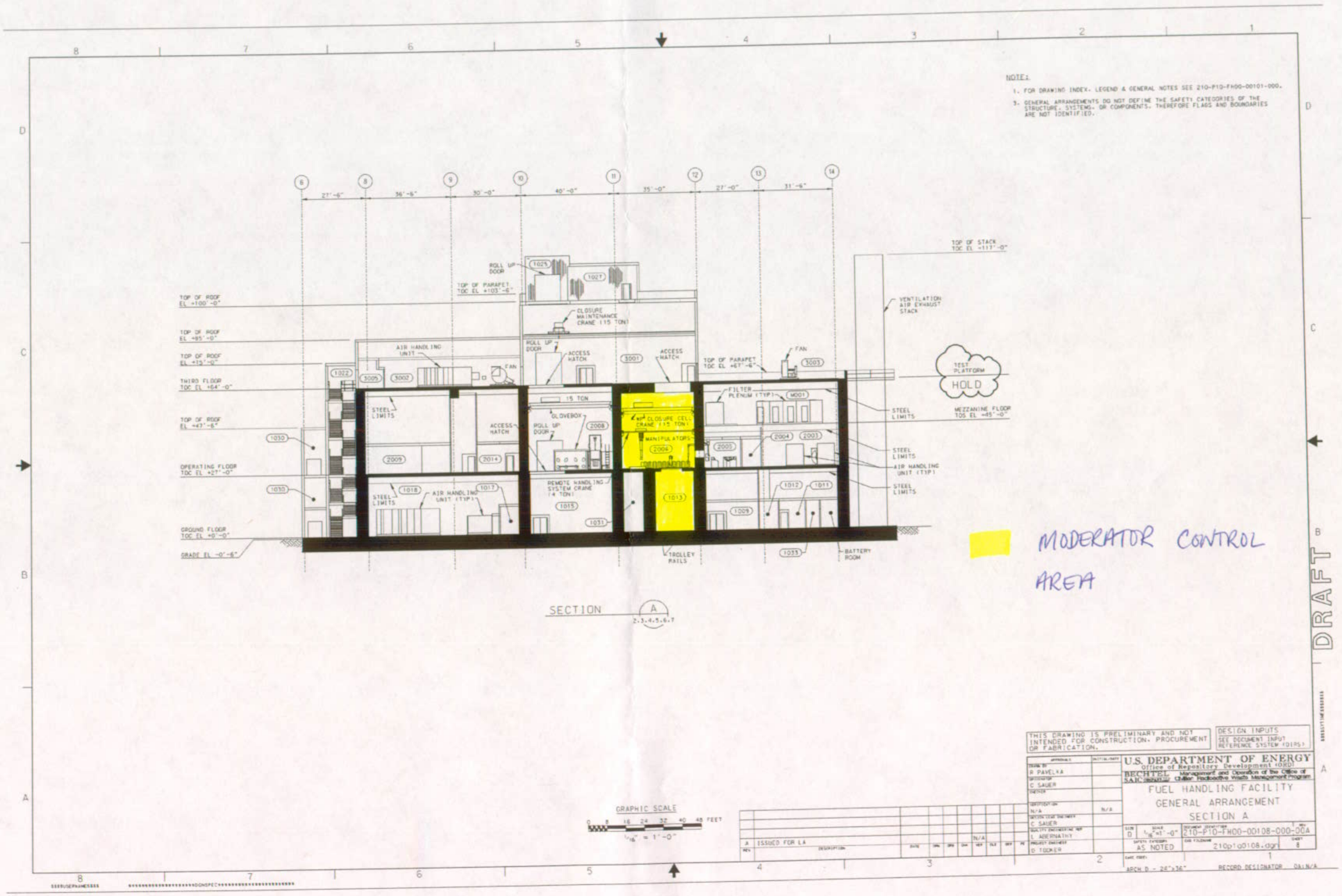
DESIGN INPUTS
U.S. DEPARTMENT OF ENERGY
BECHTEL SAIC
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
ROOF PLAN
J HERRERA
C SAUER
L ABERNATHY
D TOOKER
ISSUED FOR LA
AS NOTED
210-P10-FH00-00107-000-00A


MODERATOR CONTROL AREA
DRAFT
SECTION A
HOLD
TEST PLATFORM
TOP OF STACK TOC EL +117'-0"
VENTILATION AIR EXHAUST STACK
MEZZANINE FLOOR TOS EL +85'-0"
TOP OF ROOF EL +100'-0"
TOP OF ROOF EL +85'-0"
TOP OF ROOF EL +75'-0"
THIRD FLOOR TOC EL +64'-0"
TOP OF ROOF EL +47'-6"
OPERATING FLOOR TOC EL +27'-0"
GROUND FLOOR TOC EL +0'-0"
GRADE EL -0'-6"
TROLLEY RAILS
BATTERY ROOM
GRAPHIC SCALE
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DESIGN INPUTS
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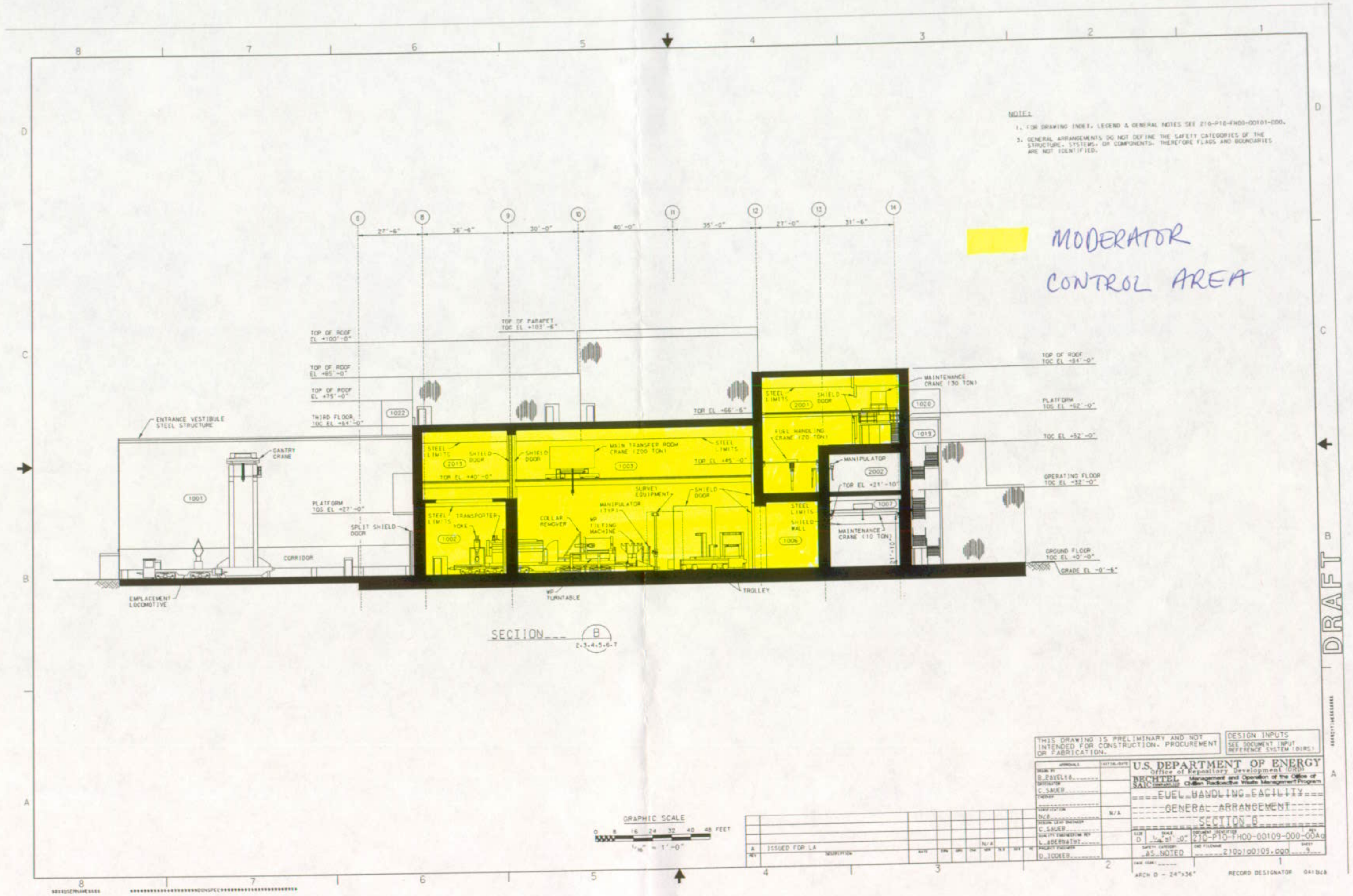
Office of Repository Development
BECHTEL SAIC
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
SECTION A
210-P10-FH00-00108-000-00A
AS NOTED
ISSUED FOR LA


MODERATOR CONTROL AREA
SECTION B
DRAFT
U.S. DEPARTMENT OF ENERGY
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
SECTION B
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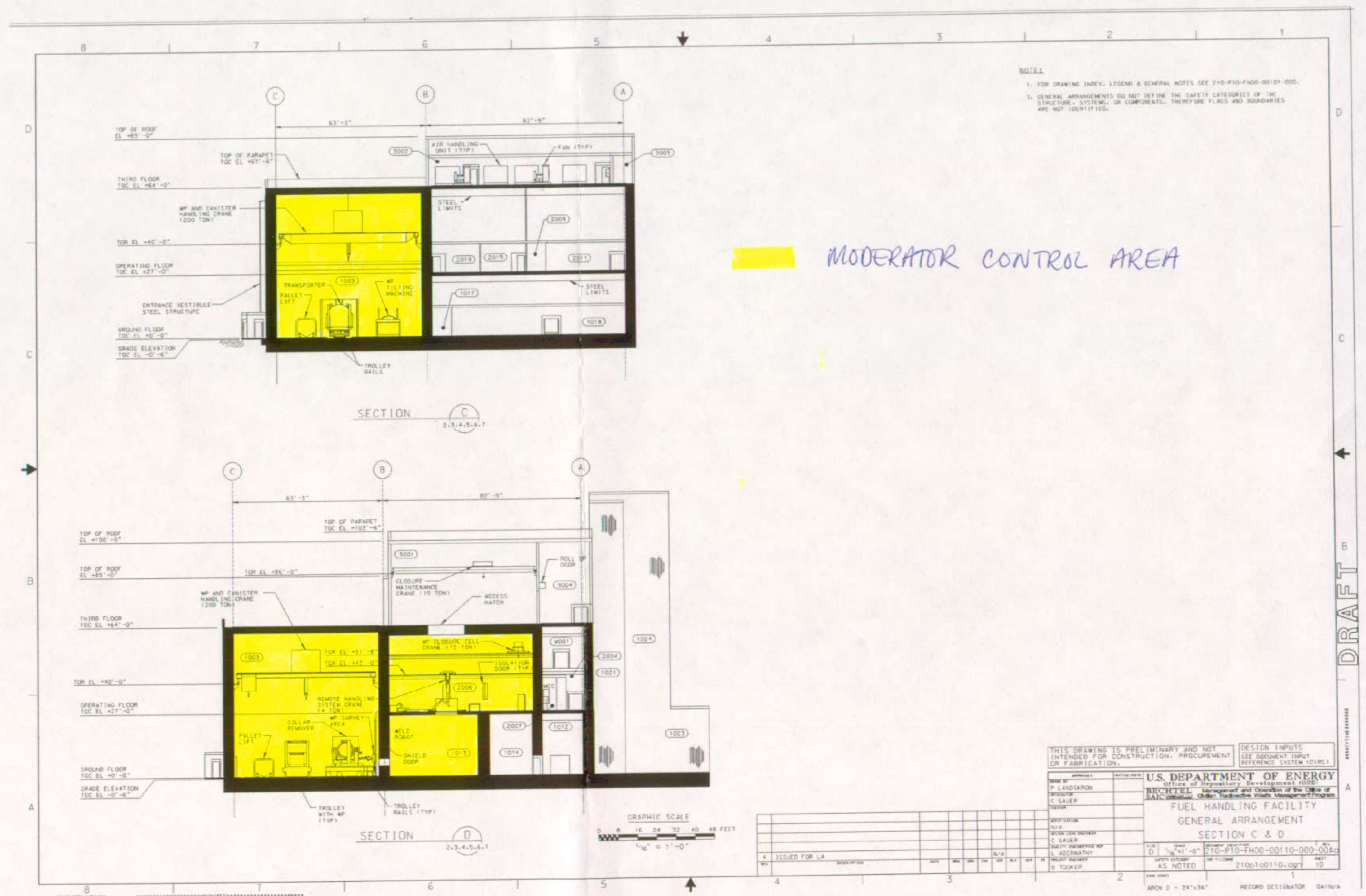
GRAPHIC SCALE


NOTES:
1. FOR DRAWING INDEX, LEGEND & GENERAL NOTES SEE 210-P10-FH00-00101-000.
2. GENERAL ARRANGEMENTS DO NOT DEFINE THE SAFETY CATEGORIES OF THE STRUCTURE, SYSTEMS, OR COMPONENTS. THEREFORE FLAGS AND BOUNDARIES ARE NOT IDENTIFIED.
MODERATOR CONTROL AREA
SECTION C
SECTION D
GRAPHIC SCALE
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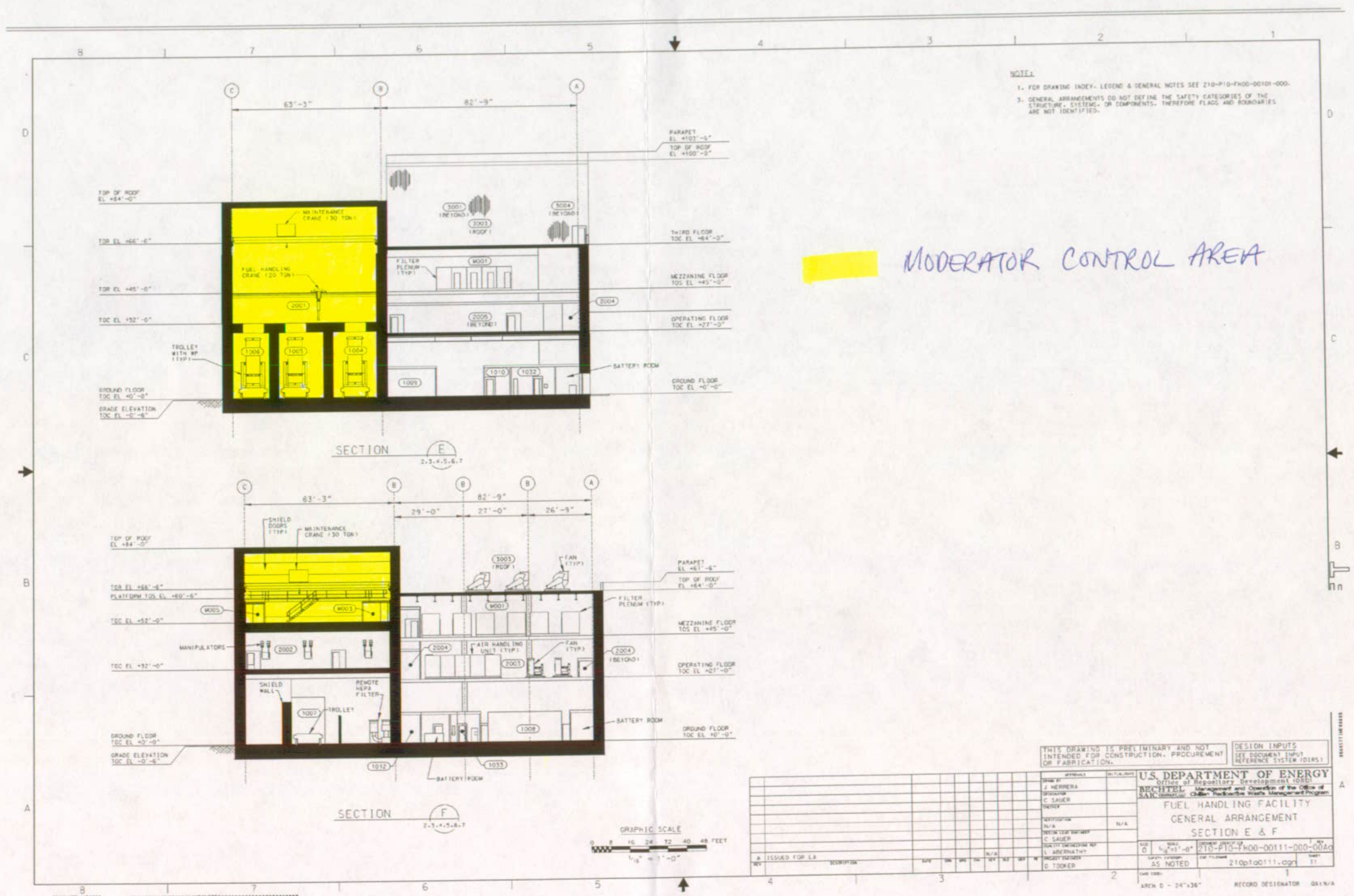
U.S. DEPARTMENT OF ENERGY
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
SECTION C & D
210-P10-FH00-00110-000-00A
ISSUED FOR LA
DRAFT


MODERATOR CONTROL AREA
NOTES:
SECTION E
SECTION F
GRAPHIC SCALE
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DESIGN INPUTS
U.S. DEPARTMENT OF ENERGY
Office of Repository Development (ORD)
BECHTEL SAIC
FUEL HANDLING FACILITY
GENERAL ARRANGEMENT
SECTION E & F
210-P10-FH00-00111-000-00A
AS NOTED
MAINTENANCE CRANE (30 TON)
FUEL HANDLING CRANE (20 TON)
TROLLEY WITH RP (TYP)
FILTER PLENUM (TYP)
BATTERY ROOM
GROUND FLOOR
OPERATING FLOOR
MEZZANINE FLOOR
THIRD FLOOR
PARAPET
SHIELD DOORS (TYP)
MANIPULATORS
SHIELD WALL
REMOTE HEPA FILTER
TROLLEY
AIR HANDLING UNIT (TYP)
FAN (TYP)
GRADE ELEVATION
63'-3"
82'-9"
29'-0"
27'-0"
26'-9"

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