U.S. DEPARTMENT OF ENERGY
OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT

NUCLEAR WASTE TECHNICAL REVIEW BOARD
FULL BOARD MEETING

SUBJECT: WASTE PACKAGE DESIGN

PRESENTER: HUGH A. BENTON

PRESENTER'S TITLE AND ORGANIZATION: MANAGER, WASTE PACKAGE DEVELOPMENT
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LAS VEGAS, NEVADA
APRIL 19-20, 1995

Topics

- Waste Package Barriers
- Current Designs
- Waste Package Costs
- Performance Analyses
- Internal Heat Transfer
- Shielding Options
- Engineering Development Plans

Functions of EBS Materials

Component	Function	Potential Contribution to Safety	
		Aggressive conditions	Less aggressive conditions
Backfill over Waste package	Protect from rock fall Divert dripping groundwater	To be evaluated	To be evaluated
Packing underneath waste package	Sorb radionuclides	To be evaluated	To be evaluated

PRE020.WP5.125.NWTRB

Functions of Barrier Materials

Component	Function	Potential Contribution to Safety	
		Aggressive conditions	Less aggressive conditions
(3rd barrier) corrosion resistant	Control microbiologically influenced corrosion	high	(not used)
2nd barrier corrosion allowance	Provide predictable corrosion rate Control radiolysis	low	moderate
1st barrier corrosion resistant	Resist corrosion under expected conditions Protect spent nuclear fuel from oxidation	high	high

PRE020.WP5.125.NWTRB

Functions of Other Container Materials

Component	Function	Potential Contribution to Safety	
		Aggressive conditions	Less aggressive conditions
Multi-purpose canister shell	Provide convenient handling at surface facilities	low	low
Filler material	Control criticality, sorb radionuclides	moderate	moderate
Fill gas	Prevent oxidation of fuel to U_3O_8 Prevent cladding oxidation	moderate	moderate
Spent nuclear fuel basket	Control criticality, conduct heat	high	high

Function of Waste Form: Spent Nuclear Fuel

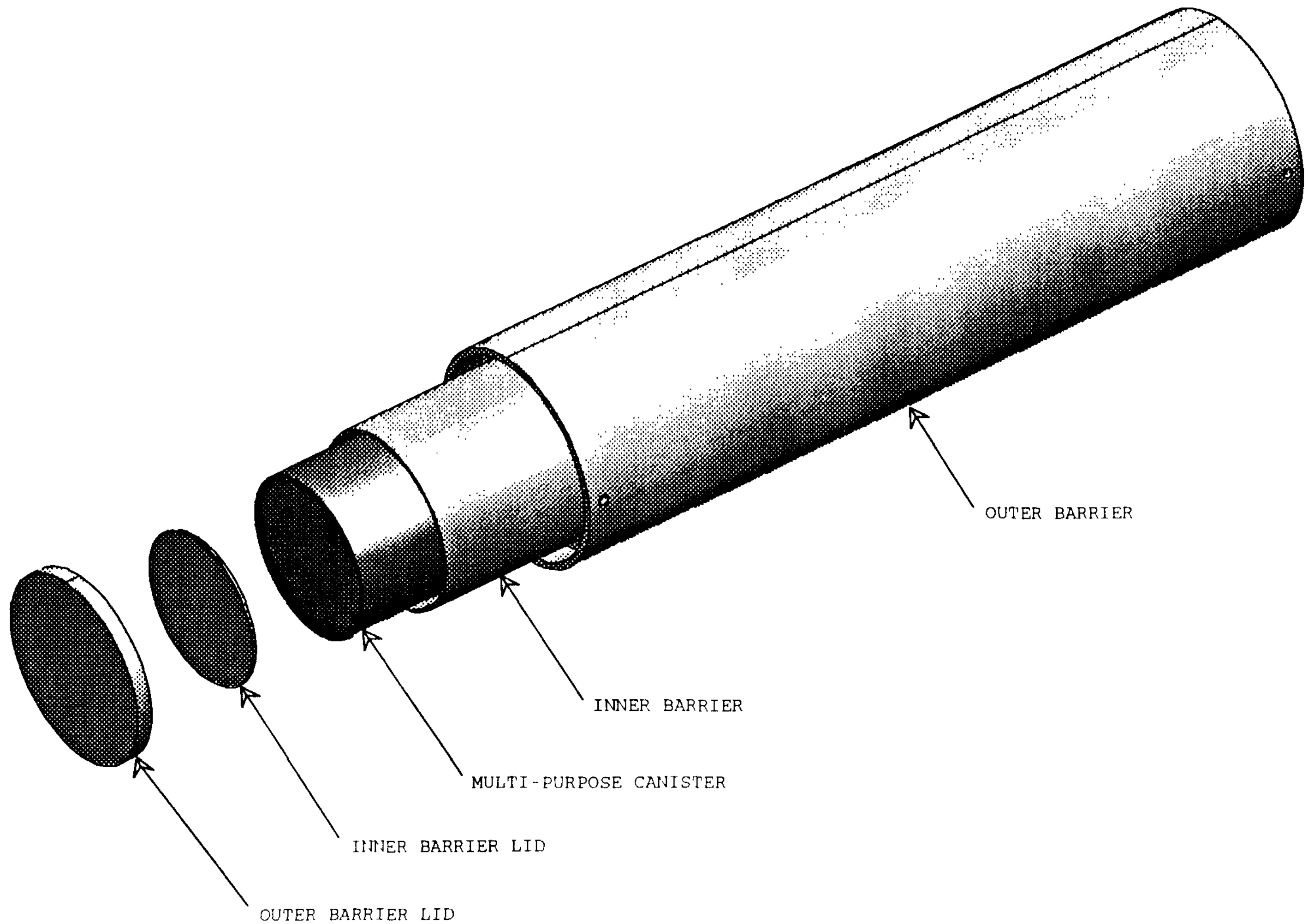
Component	Function	Potential Contribution to Safety	
		Aggressive conditions	Less aggressive conditions
Spent nuclear fuel cladding	Protect UO_2 from oxidation Limit access of water to UO_2	high	high
Spent nuclear fuel oxide	Limit solubility of radionuclides in water	moderate	moderate

PRE020.WP5.125.NWTRB

Function of Waste Form: High-Level Waste

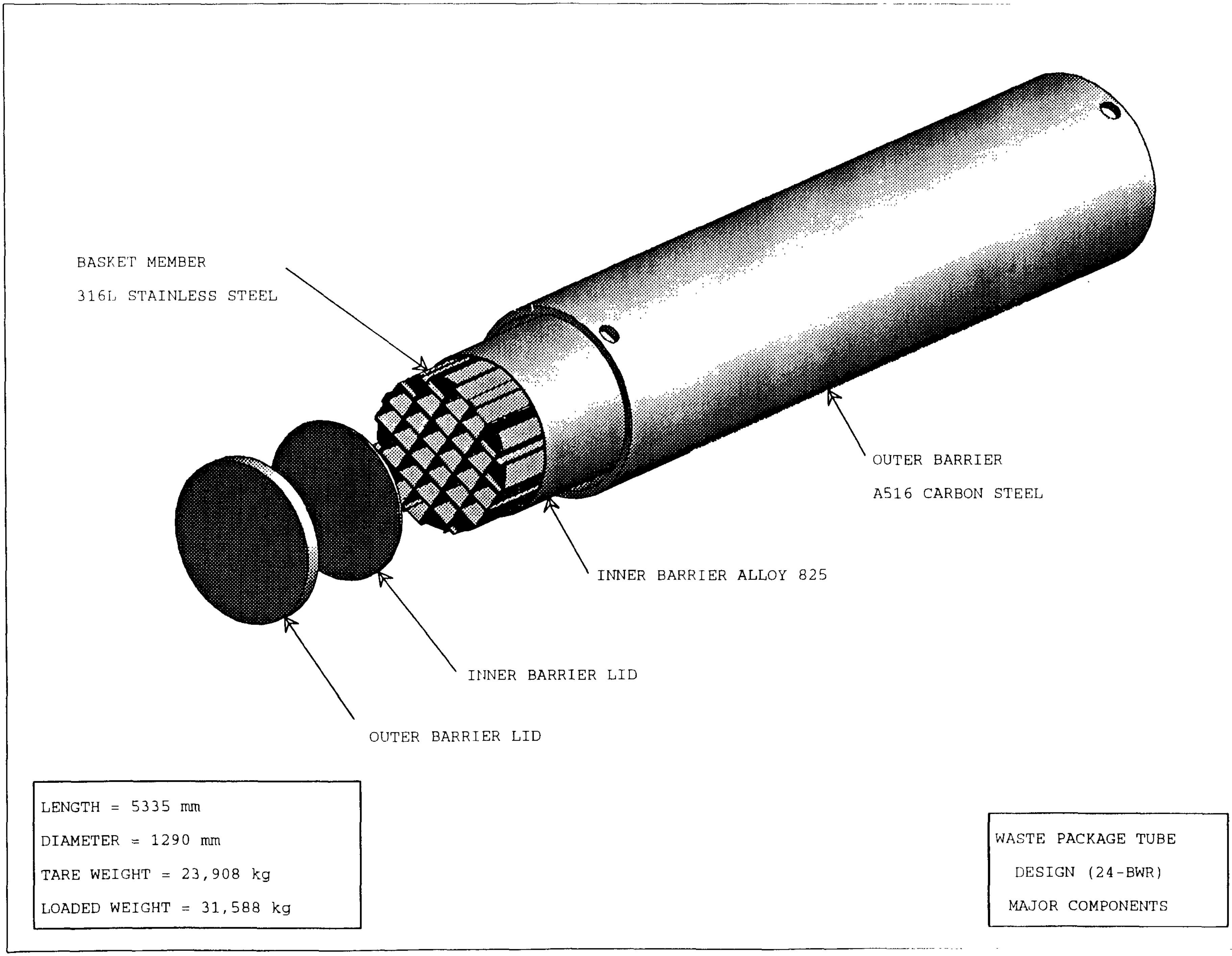
Component	Function	Potential Contribution to Safety	
		Aggressive conditions	Less aggressive conditions
Pour canister	Allow handling at surface facilities	low	low
High-level waste glass	Limit solubility of radionuclides in water	moderate	moderate

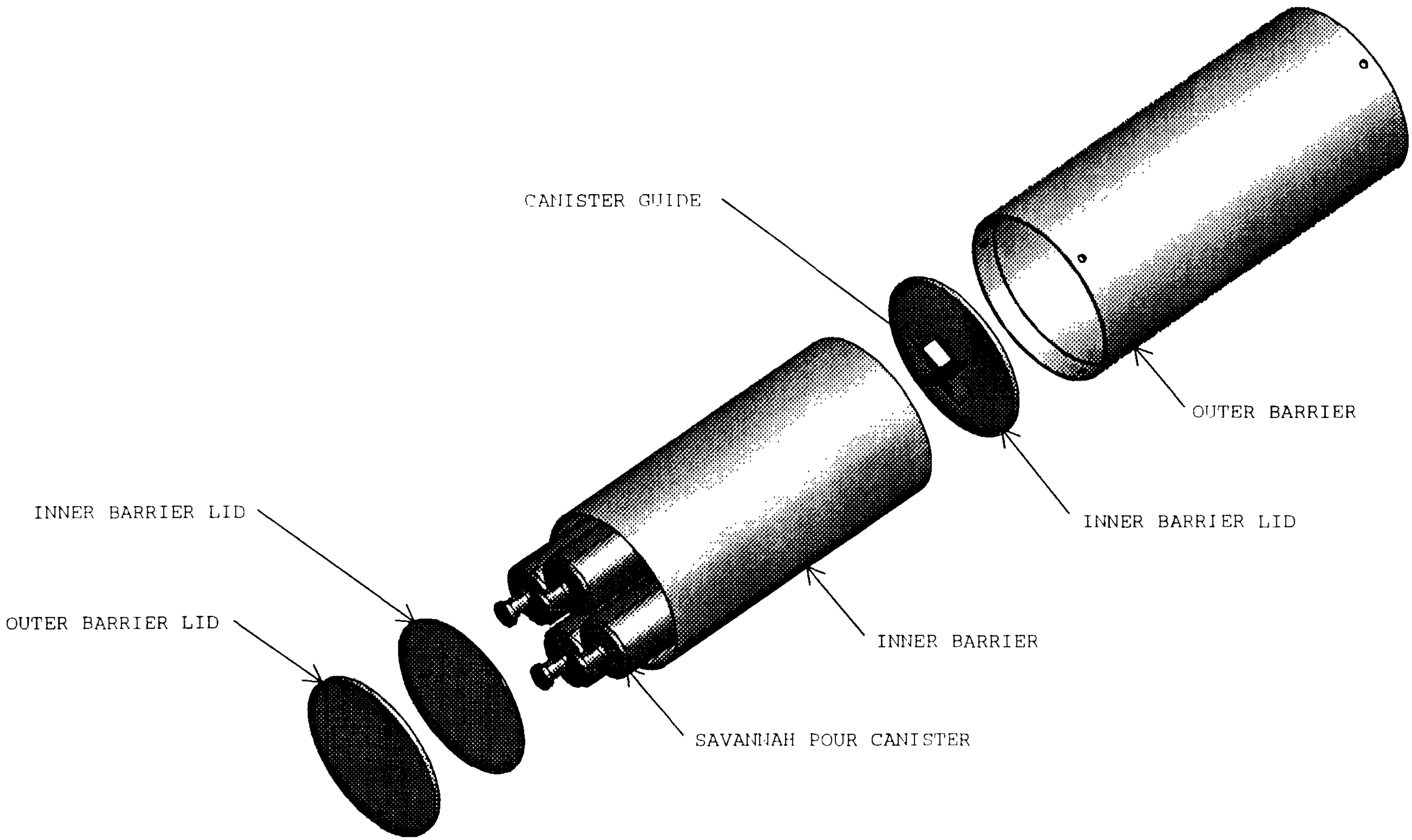
PRE020.WP5.125.NWTRB



LENGTH = 5682 mm
DIAMETER = 1802 mm
TARE WEIGHT = 31,176 kg
LOADED WEIGHT = 65,900 kg (PWR)
LOADED WEIGHT = 65,463 kg (BWR)

MULTI-PURPOSE CANISTER DISPOSAL CONTAINER
(21-PWR/40-BWR)


BASKET MEMBER
316L STAINLESS STEEL
OUTER BARRIER
A516 CARBON STEEL
INNER BARRIER ALLOY 825
INNER BARRIER LID
OUTER BARRIER LID
LENGTH = 5335 mm
DIAMETER = 1290 mm
TARE WEIGHT = 23,908 kg
LOADED WEIGHT = 31,588 kg
WASTE PACKAGE TUBE
DESIGN (24-BWR)
MAJOR COMPONENTS



LENGTH - 3680 mm

DIAMETER = 1709 mm

TARE WEIGHT = 13,494 kg

LOADED WEIGHT = 22,222 kg

DEFENSE HIGH LEVEL WASTE DISPOSAL CONTAINER

Waste Package Barrier Materials

		Current Design		Alternate Design	
Container for	Barrier	Aggressive	Less Aggressive	Aggressive	Less Aggressive
Multi-Purpose Canister and Uncanistered Fuel	Outer Middle Inner	Monel 400 A 516 825	A 516 825	Ceramic* 21/4 Cr-1Mo C-22	2 1/4 Cr-1Mo C-22
Defense High Level Waste	Outer Middle Inner Filler	Monel 400 70/30 Cu-Ni 825	70/30 Cu-Ni 825	Monel 400 70/30 Cu-Ni C-22	70/30 Cu-Ni 825 C-22

* 70/30 Cu-Ni

WASTE PACKAGE
PERCENT OF COST BY CATEGORY
21 PWR UNCANISTERED FUEL TUBE TYPE

COST CATEGORY	PERCENT OF WASTE PACKAGE COST
Material	69
Labor	13
Weld material	1
Other	17

PRE020.WP5.125.NWTRB

PRELIMINARY PREDECISIONAL DRAFT

Cost of Waste Package Materials

Material	Component	$/kg
A 516 Carbon Steel	Outer Barrier	$1.57
316L Stainless Steel	Basket Structure	3.20
6063 Aluminum	Thermal Shunt	6.61
C71500 Cupronickel (70/30)	Outer Barrier	11.46
Alloy 825	Inner Barrier	12.43
316L SS - Boron Alloy (316B6A)	Basket	28.66

Preliminary Predecisional Draft

Total Cost of Disposal Containers and Uncanistered Fuel Waste Packages

(With Contingency)

Disposal Containers and Uncanistered Fuel Waste Package	Number of Units	Unit Cost $ (thousands)	Total Cost $ (millions)
Large MPC Disposal Container: 21 PWR/40 BWR	6289	$190	$1,195
Small MPC Disposal Container: 12 PWR/24 BWR	2281	171	390
DHLW Disposal Container (4-canisters)	3362	284	955
21 PWR Uncanistered Fuel Waste Package (includes basket)	183	379	69
Performance Confirmation (assume 21 PWR)	5	379	2
Totals	12,120		$2,611

Preliminary Predecisional Draft

Confinement by Cladding

- Intact cladding prevents release from UO_2

- Even perforated cladding (small pinhole) confines fuel and limits access by water

Conservative Approach to Evaluation

- Assume that radiolysis will convert atmosphere into aggressive species

- Assume that degradation is limited only by elemental availability

Degradation by Oxidation

- Occurs only after breach of disposal container
- Degradation mechanisms
  - General oxidation of cladding (all fuel)
  - Fuel oxidation to U_3O_8 and subsequent splitting of cladding (perforated cladding only)

Calculation of Amount of Degradation

- Fuel characteristics: 42 GWd/MTU, 22 years old, 3.92% enrichment
- Hottest rod in a 21 PWR multi-purpose canister, no backfill
- 4.3 m drift diameter
- 16 m waste package spacing
- Rate of degradation, as reported by Einziger (IHLRWMC 1994, p. 554)

Cladding Oxidation

- Worst-case conditions chosen
  - High thermal load (20.5 kg U/m^2 = 83 MTU/acre)
  - No protection by disposal container
- Only 2.3% of cladding thickness oxidizes in first 1000 years
- After 1000 years, cladding temperature is low; oxidation rate is negligible

Fuel Oxidation (Perforated Cladding Only)

- Considered both high and low thermal loads
  - 20.5 kg U/m^2 = 83 MTU/acre
  - 5.93 kg U/m^2 = 24 MTU/acre
- If container remains intact for even 200 years, splitting will not occur in first 10,000 years

Significance of Oxidation

- General oxidation of cladding is negligible
- <u>Perforated</u> cladding requires inert atmosphere to protect UO_2, but only during the period when container is too hot for aqueous corrosion

Specification for Multi-Purpose Canister Water Content

- "the residual water content of the multi-purpose canister (shall be) less than 0.25 volume percent"
- Multi-purpose canister is to be evacuated to pressure of no more than 300 Pa for 10 minutes
- But, vapor pressure of liquid water is at least 2600 Pa
- So, water is in vapor phase, amount of water is no more than 13 g

Potential Effects of Atmosphere in Multi-Purpose Canister

- Hydrogen embrittlement of fuel cladding
- Oxidation of cladding or multi-purpose canister interior
- Corrosion by radiolytically produced nitric acid

Hydrogen Embrittlement of Fuel Cladding

- For 21 PWR multi-purpose canister, hydrogen increase in cladding cannot exceed 0.6 ppm
- New cladding contains up to 25 ppm hydrogen
- Irradiated cladding may reach 500 to 600 ppm hydrogen
- Hydrogen embrittlement in multi-purpose canister is negligible

Oxidation of Cladding in Multi-Purpose Canister

- Total amount of oxygen in multi-purpose canister cannot exceed 15 g (11 g from water; 4 g from residual air)
- Oxidation of shell and basket is limited to 45 nm
- Oxidation of fuel cladding is limited to 11 nm
- Oxidation in multi-purpose canister is negligible

Corrosion by Nitric Acid (HNO_3) (Bulk Condensation)

- Oxygen supply limits partial pressure of HNO_3 to 170 Pa
- Vapor pressure of liquid 90% HNO_3 is 3,000 Pa at 22°C
- Vapor pressure increases rapidly with increasing temperature
- No condensation of HNO_3 can occur

Corrosion by Nitric Acid (Water Films)

- Even without radiolysis, relative humidity at 22°C is no larger than 15%
- Any radiolysis will reduce humidity
- Water-film corrosion normally requires relative humidities of 60% or higher
- No water-film corrosion can occur

Conclusions

- Specification for allowable amount of water was evaluated
- Considered potential for effects of
  - Hydrogen embrittlement
  - Oxidation
  - Nitric acid formation
- Even with very conservative treatment, effects of atmosphere (including water vapor) are negligible

Factors Affecting Thermal Response

- **Far-field temperatures depend on**

  Area mass loading (MTU/acre)

- **Near-field temperatures depend on**

  Far-field temperatures

  Waste package spacing

  Spent nuclear fuel age

  Emplacement drift diameter

  Waste package time-dependent heat output

Factors Affecting Thermal Response
(Continued)

- Waste package internal temperatures depend on

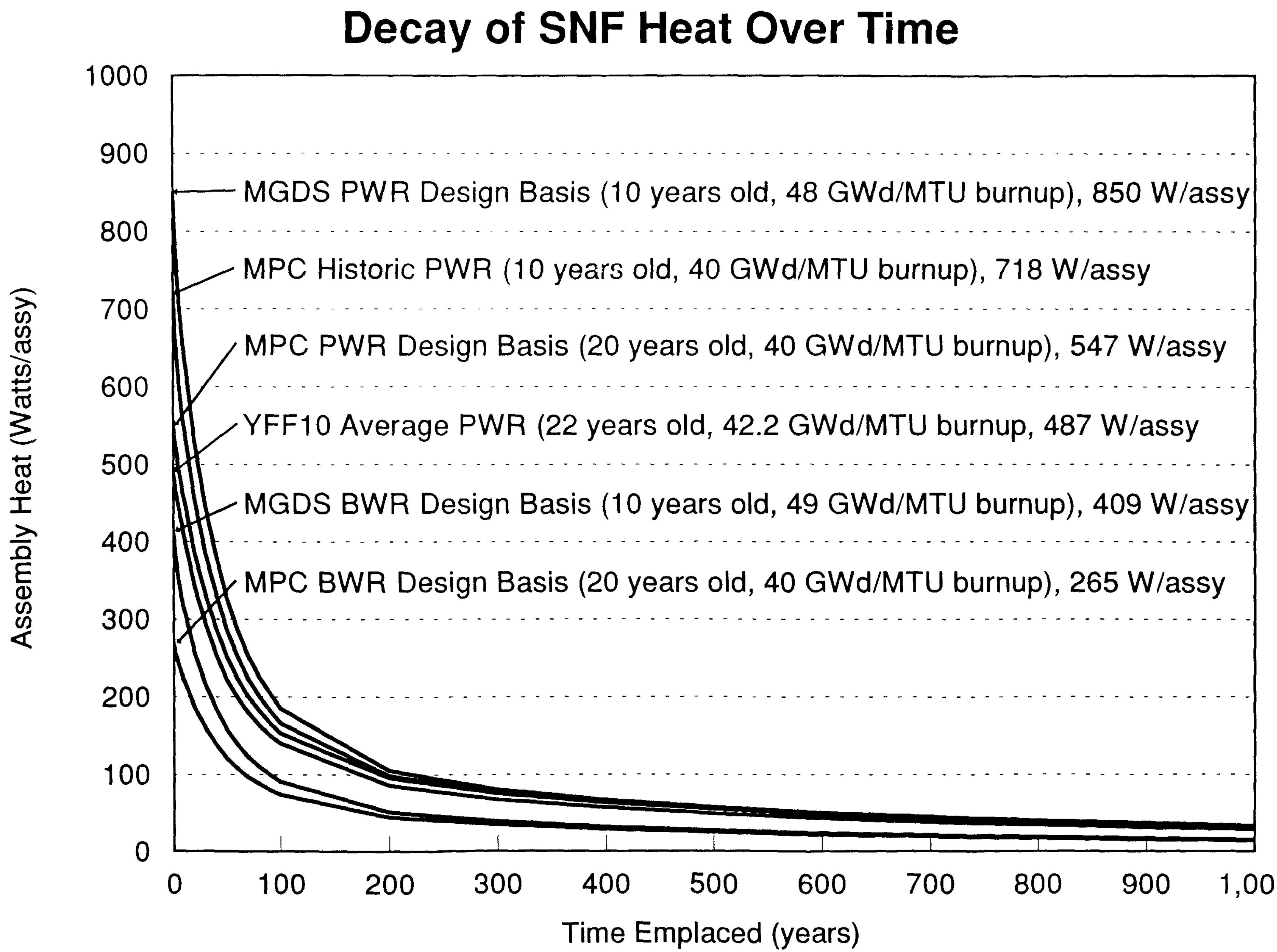
  Near-field temperatures

  Spent nuclear fuel characteristics

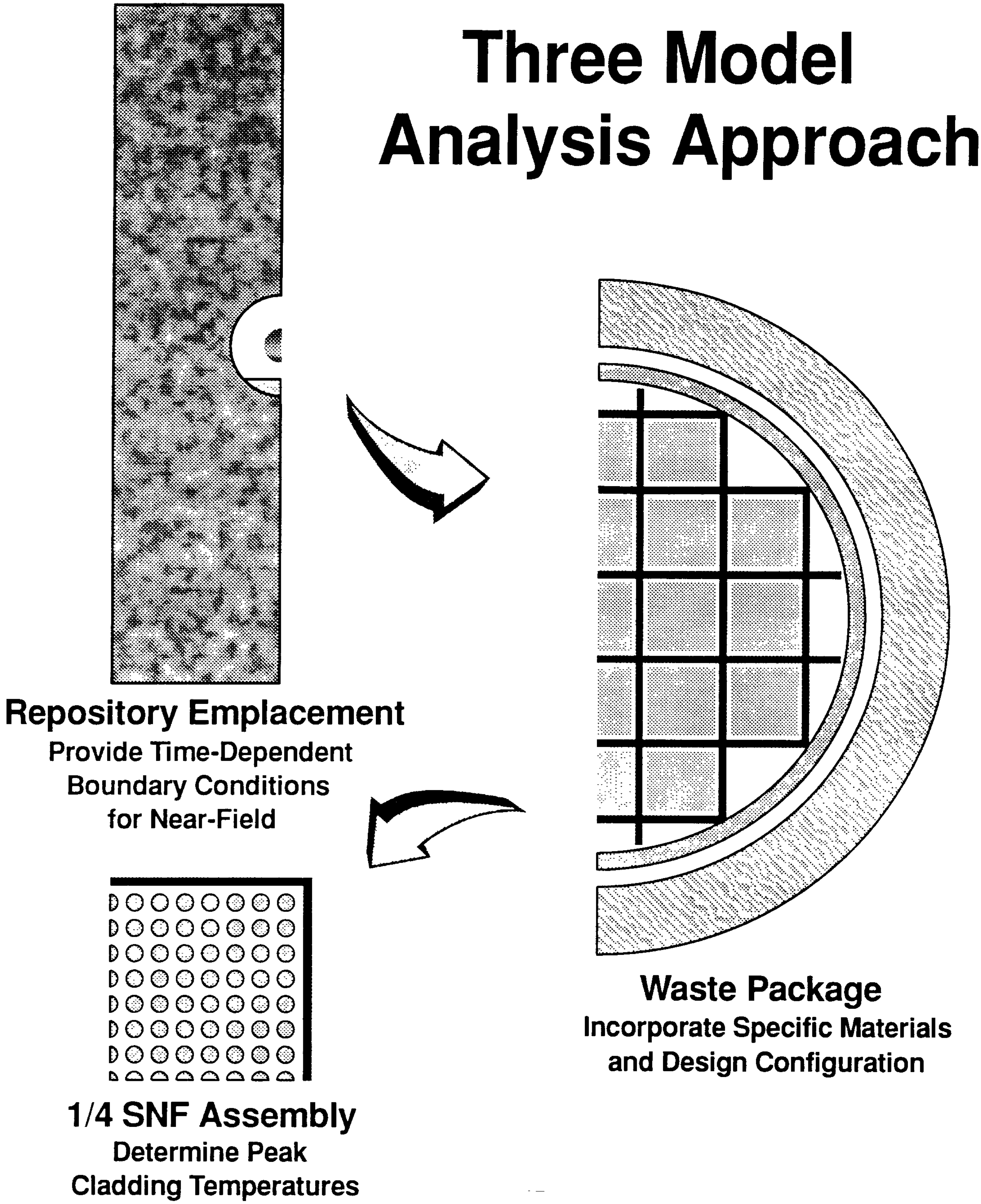
  Number of assemblies

  Materials of fabrication

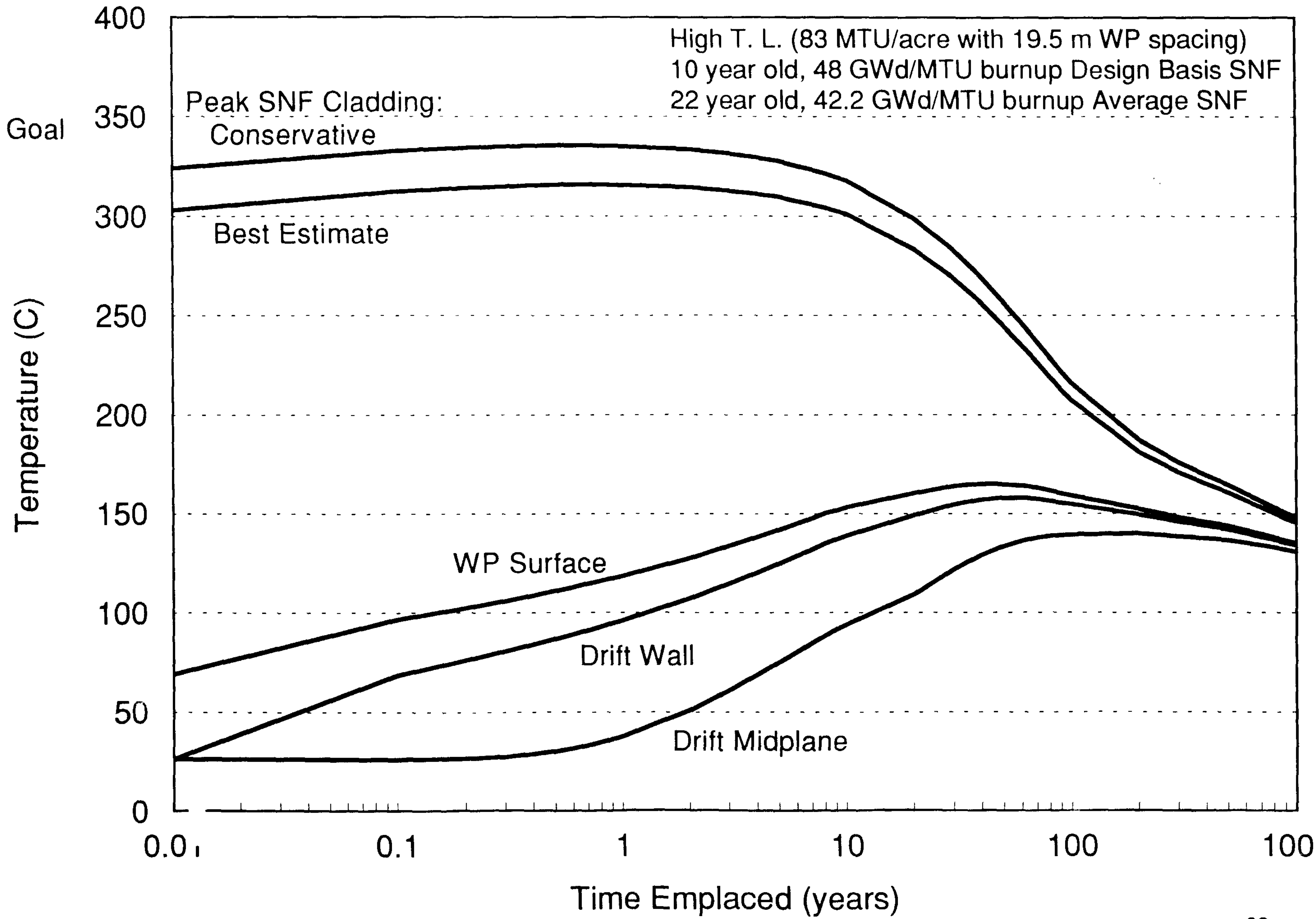
  Design type (flux trap, burnup credit)

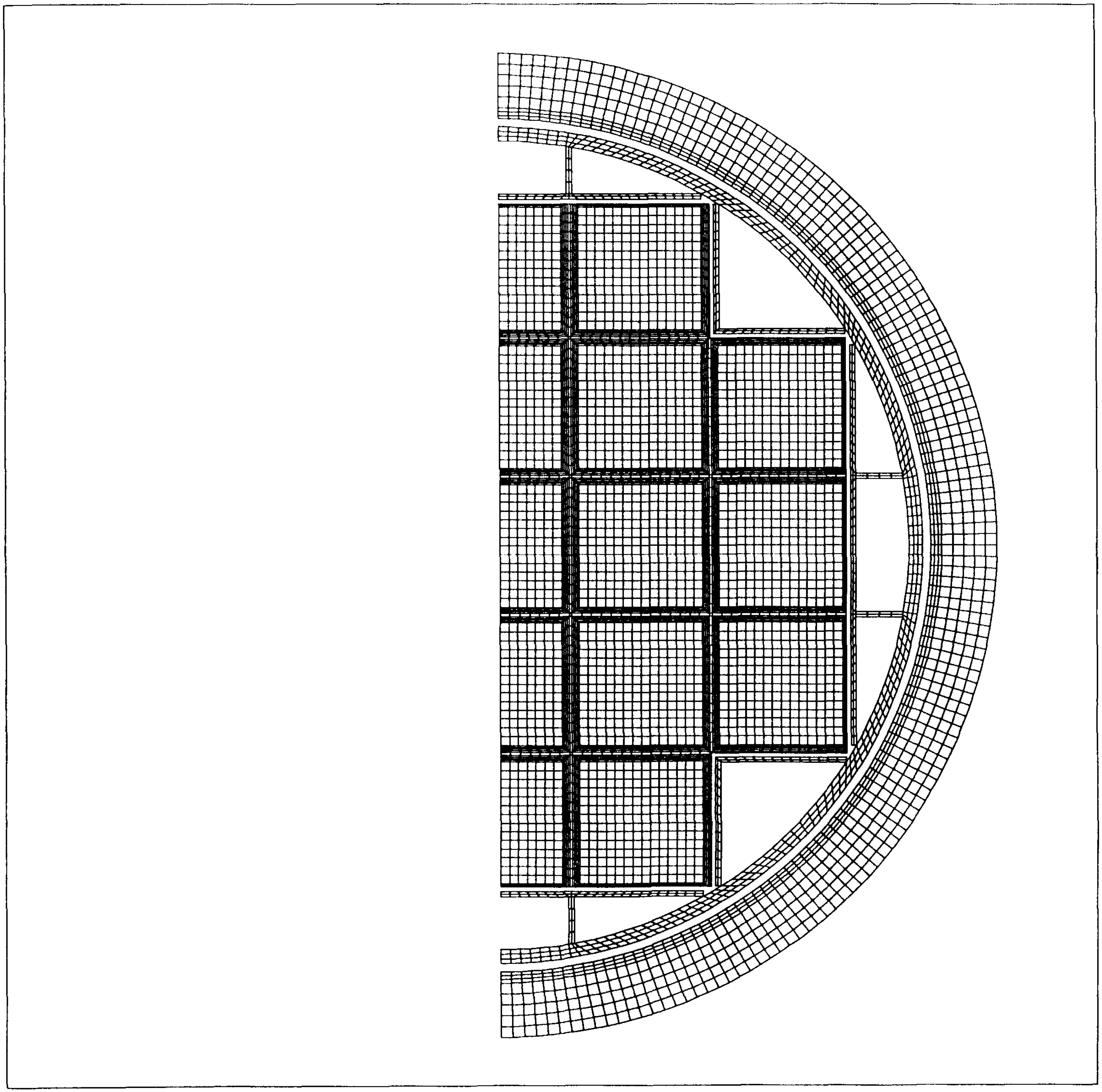

Decay of SNF Heat Over Time
1000
900
800
700
600
500
400
300
200
100
0
Assembly Heat (Watts/assy)
MGDS PWR Design Basis (10 years old, 48 GWd/MTU burnup), 850 W/assy
MPC Historic PWR (10 years old, 40 GWd/MTU burnup), 718 W/assy
MPC PWR Design Basis (20 years old, 40 GWd/MTU burnup), 547 W/assy
YFF10 Average PWR (22 years old, 42.2 GWd/MTU burnup, 487 W/assy
MGDS BWR Design Basis (10 years old, 49 GWd/MTU burnup), 409 W/assy
MPC BWR Design Basis (20 years old, 40 GWd/MTU burnup), 265 W/assy
0
100
200
300
400
500
600
700
800
900
1,00
Time Emplaced (years)

Three Model Analysis Approach



21 PWR MPC Temperatures

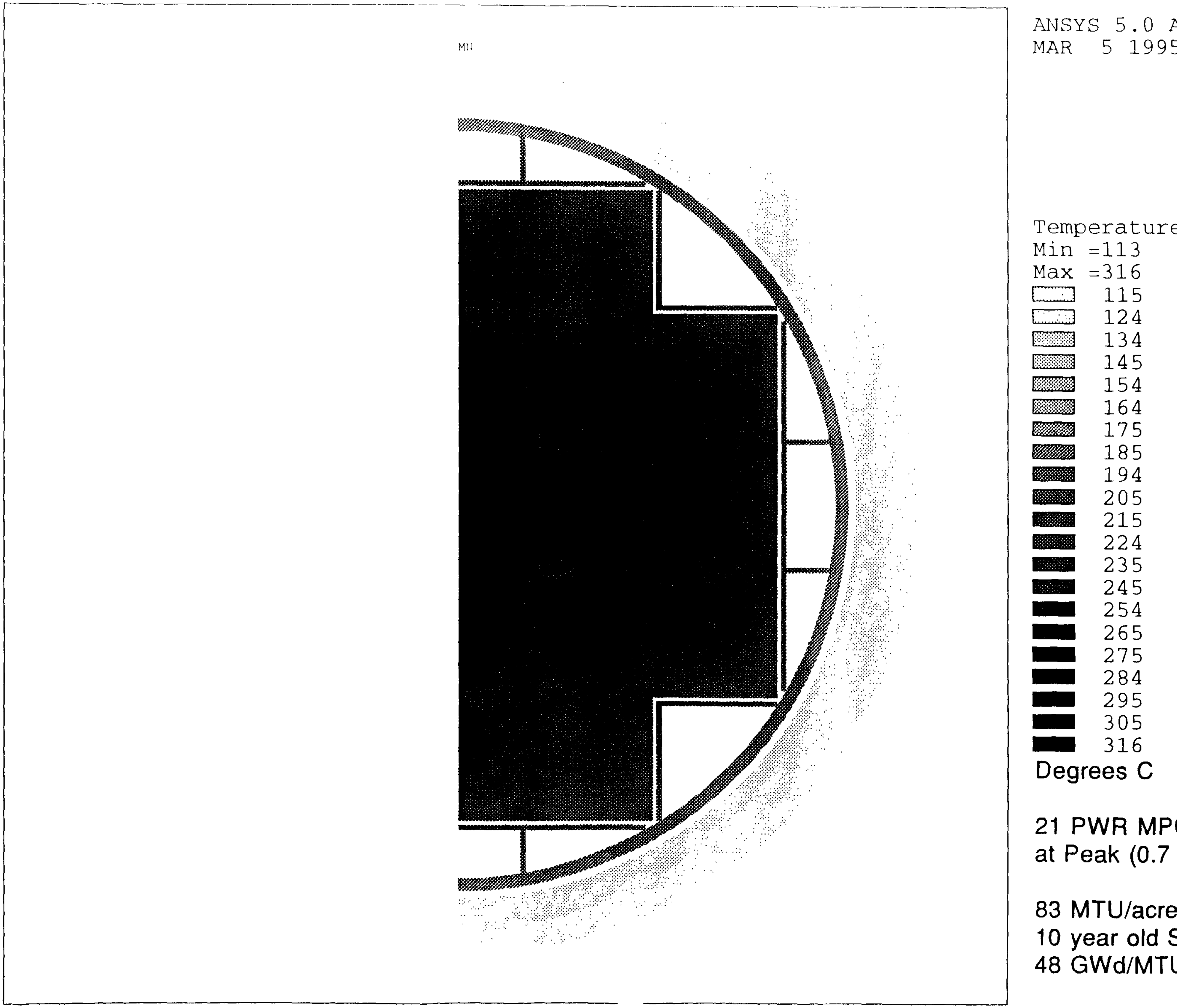




ANSYS 5.0 A
MAR 4 1995

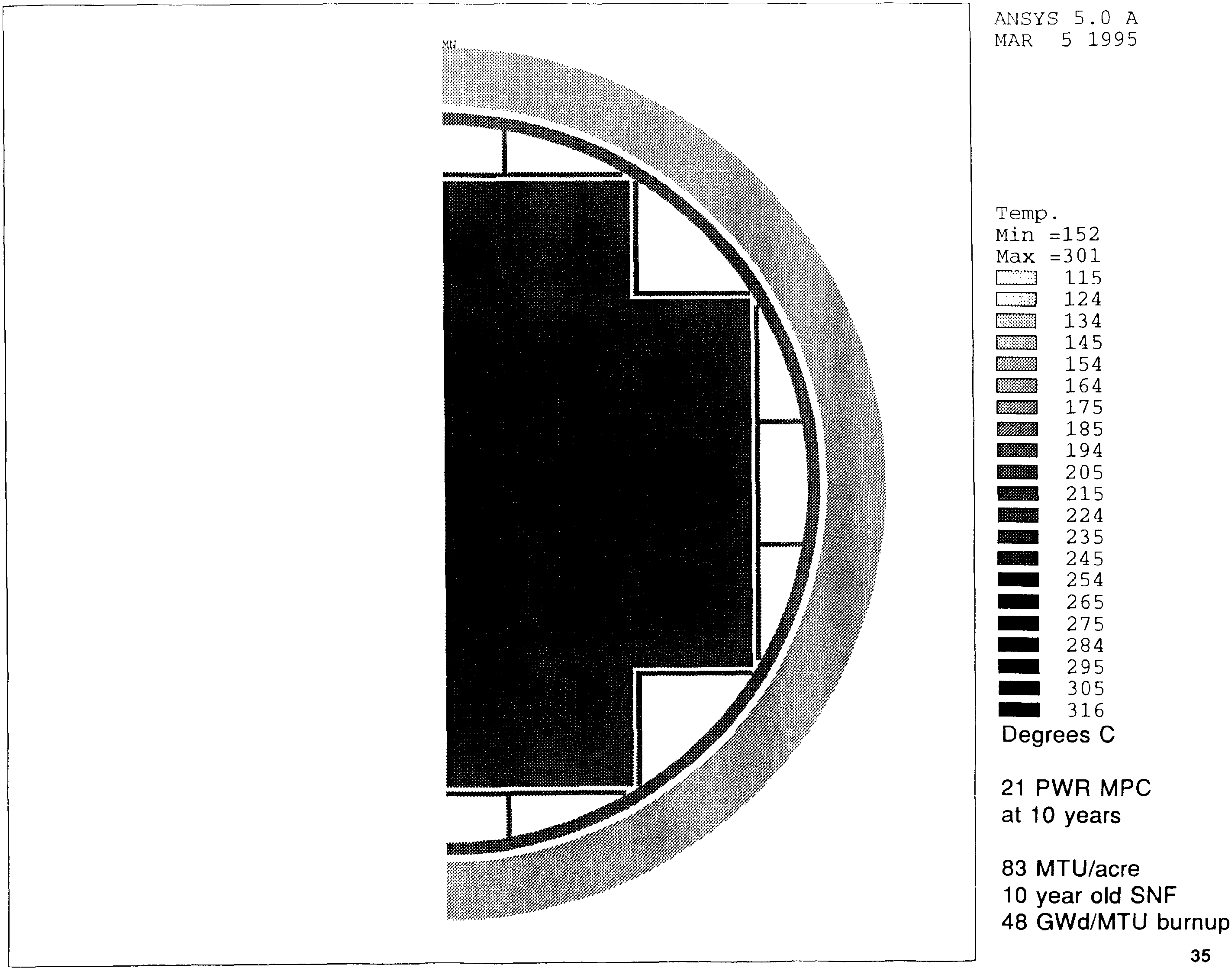
Finite-Element Model

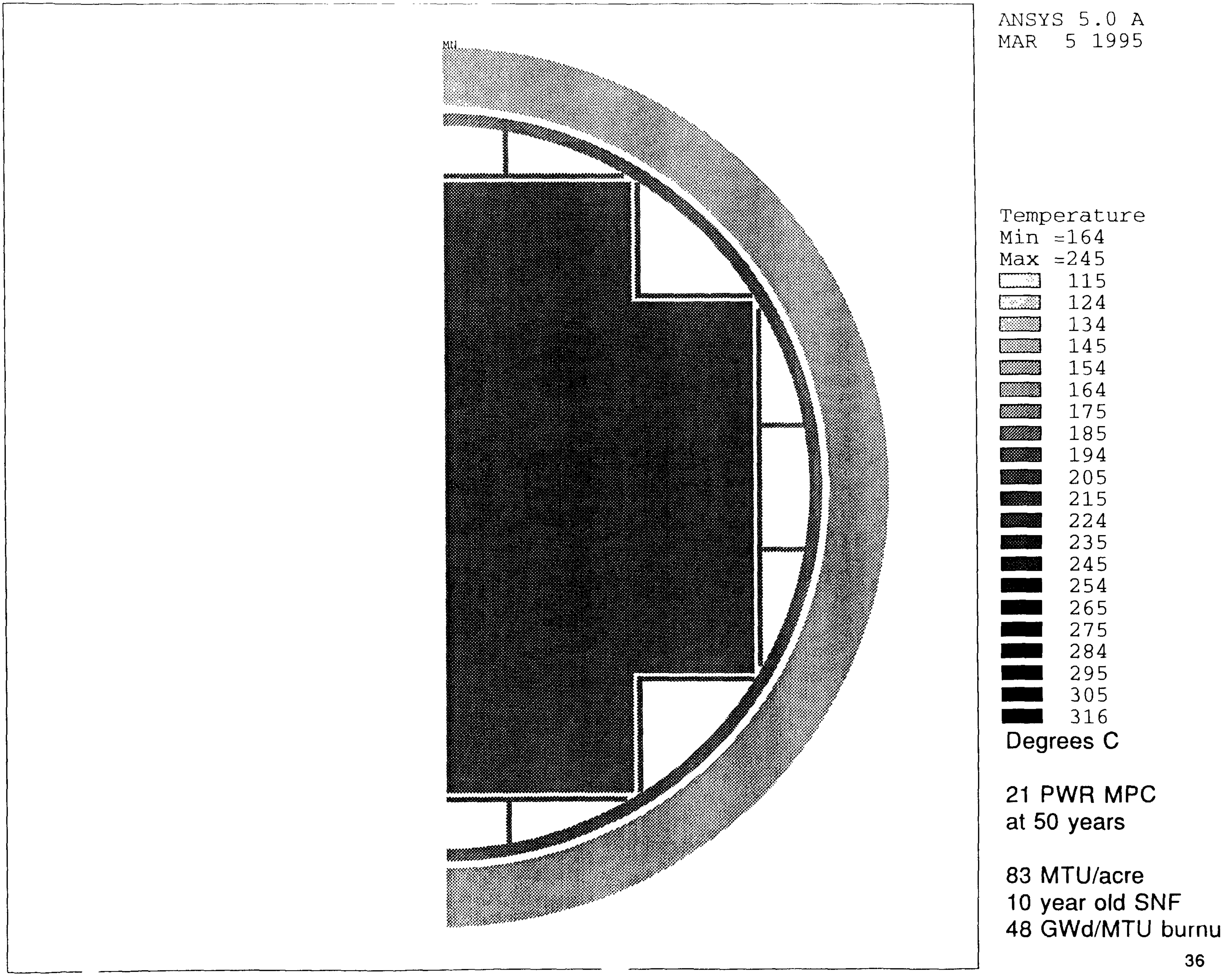
21 PWR MPC
Conceptual Design
w/ Disposal Container



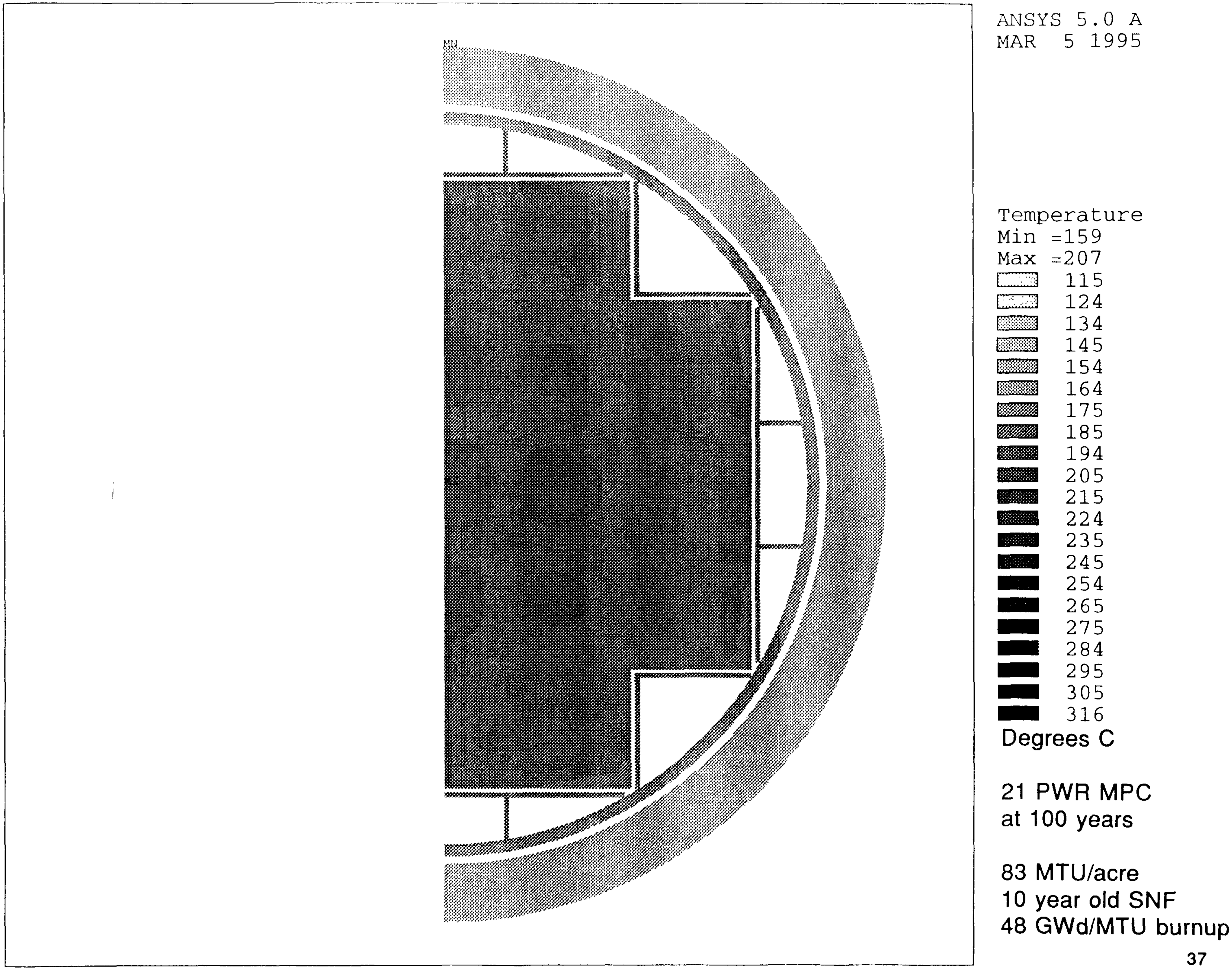
21 PWR MPC
at Peak (0.7 years)

83 MTU/acre
10 year old SNF
48 GWd/MTU burnu


MN
ANSYS 5.0 A
MAR 5 1995
Temp.
Min =152
Max =301
115
124
134
145
154
164
175
185
194
205
215
224
235
245
254
265
275
284
295
305
316
Degrees C
21 PWR MPC
at 10 years
83 MTU/acre
10 year old SNF
48 GWd/MTU burnup



ANSYS 5.0 A
MAR 5 1995
MN
Temperature
Min =164
Max =245
115
124
134
145
154
164
175
185
194
205
215
224
235
245
254
265
275
284
295
305
316
Degrees C
21 PWR MPC
at 50 years
83 MTU/acre
10 year old SNF
48 GWd/MTU burnu


MN
ANSYS 5.0 A
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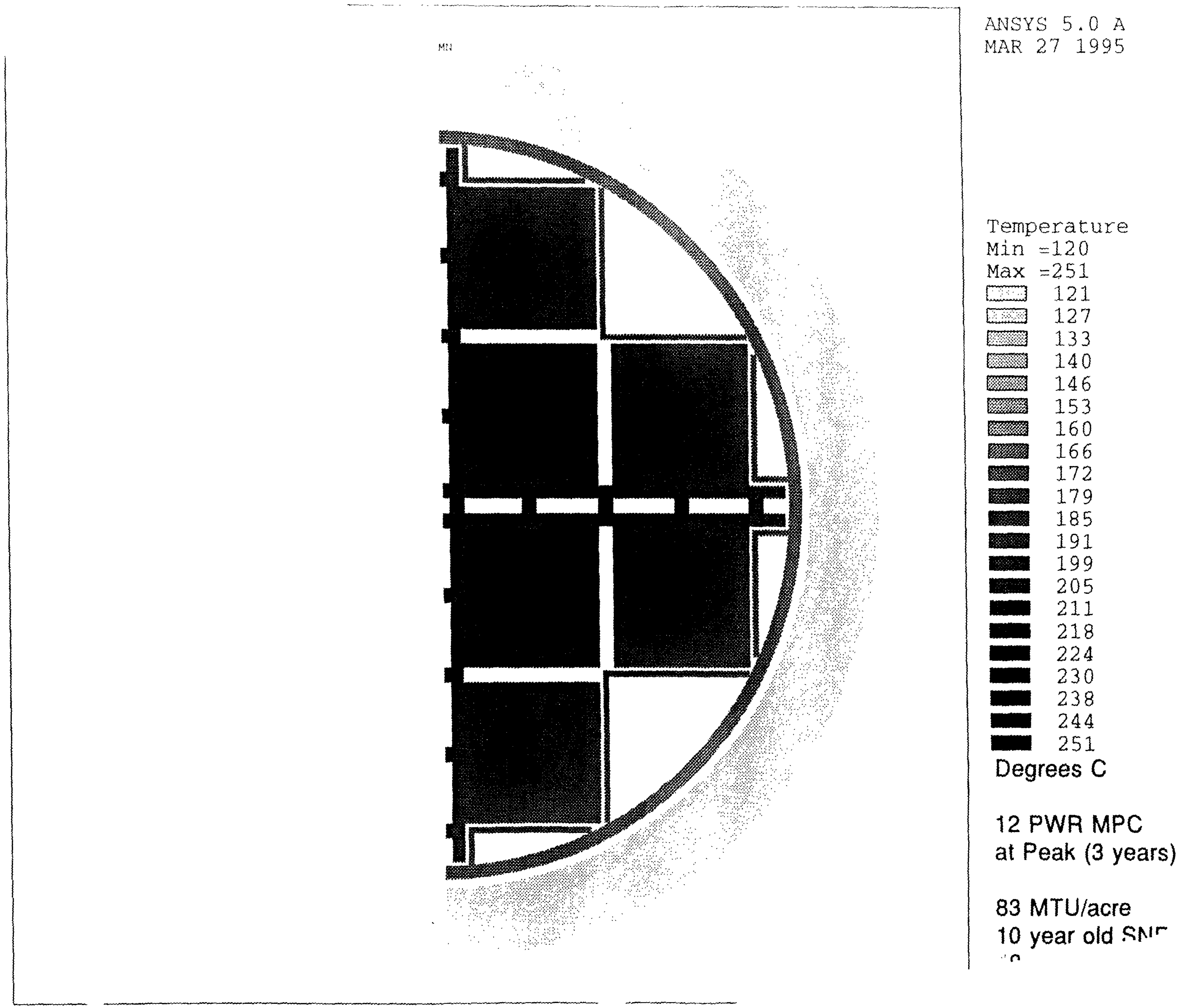
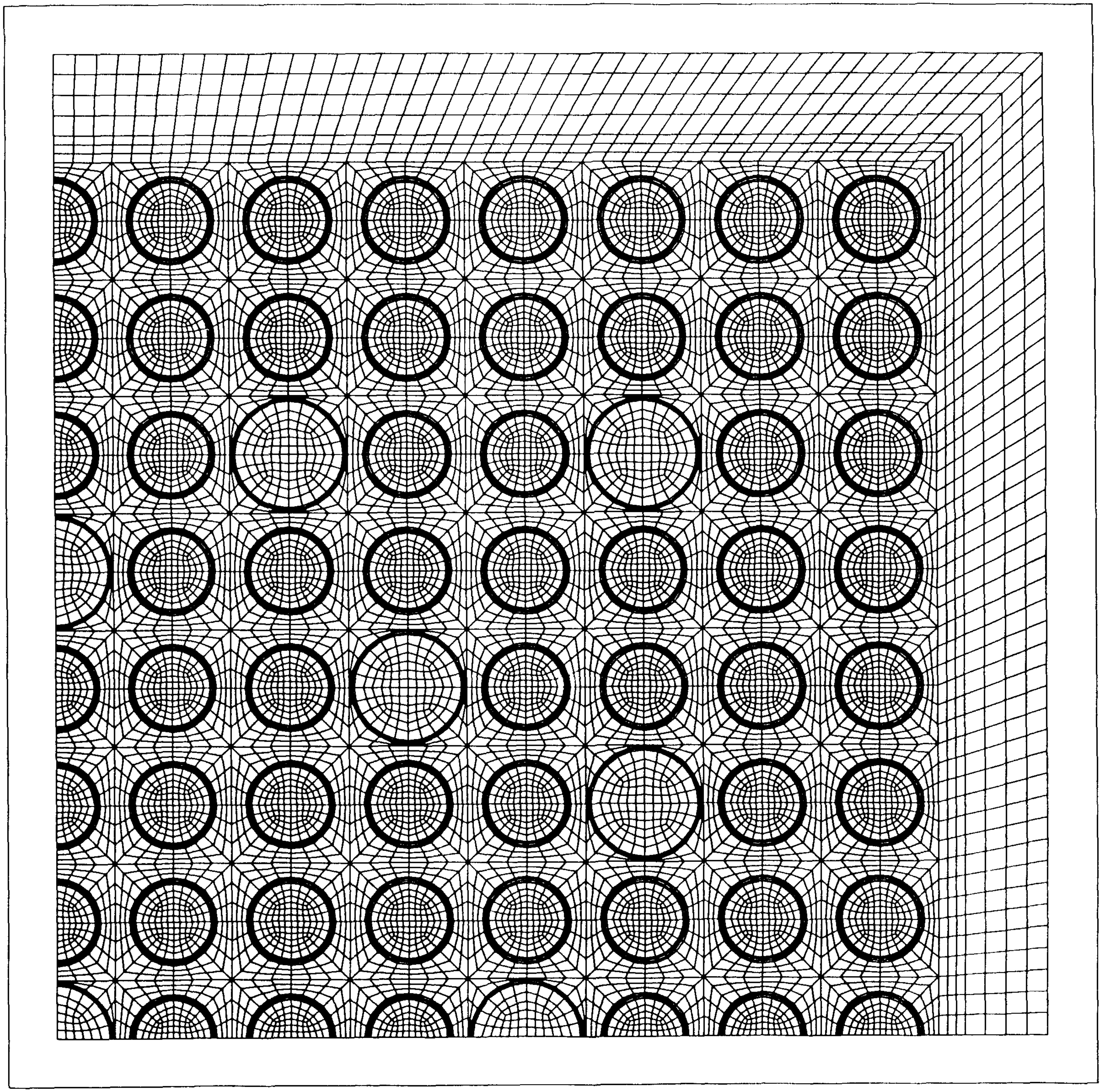
Temperature
Min =159
Max =207
115
124
134
145
154
164
175
185
194
205
215
224
235
245
254
265
275
284
295
305
316
Degrees C
21 PWR MPC
at 100 years
83 MTU/acre
10 year old SNF
48 GWd/MTU burnup

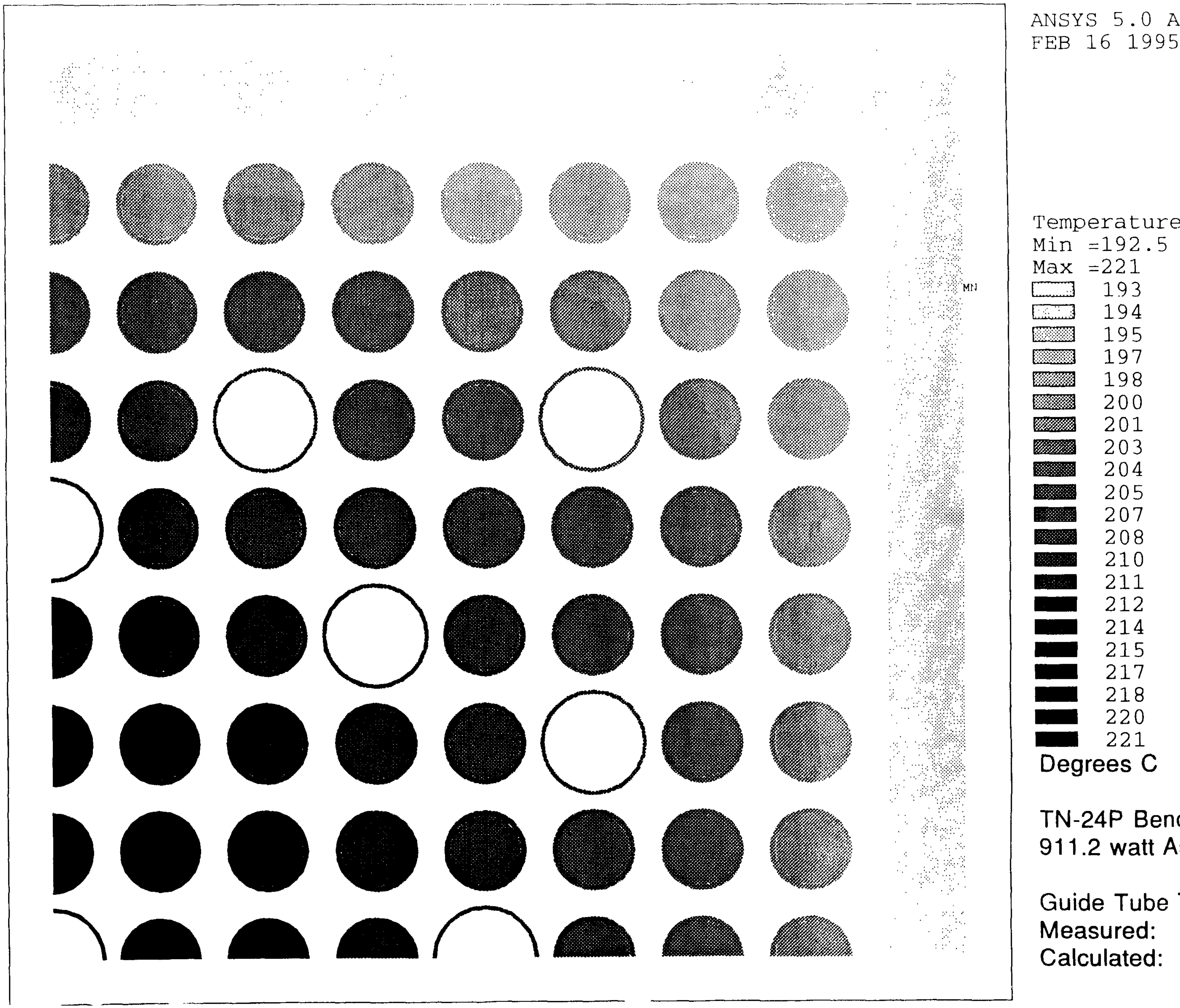

ANSYS 5.0 A
MAR 27 1995
Temperature
Min =120
Max =251
121
127
133
140
146
153
160
166
172
179
185
191
199
205
211
218
224
230
238
244
251
Degrees C
12 PWR MPC
at Peak (3 years)
83 MTU/acre
10 year old SNF



Finite-Element Model

Westinghouse 15x15
PWR SNF Assembly

Benchmark Evaluation
Against TN-24P Tests



TN-24P Benchmark
911.2 watt Assembly

Guide Tube Temps.
Measured: 206.2 C
Calculated: 214.6 C

In-Repository Radiation Shielding

- Reasons for shielding
  - Limit worker exposure
  - Protect materials (radiolysis)

Repository Radiation Shielding
(Continued)

- Requirements
  - Title 10 CFR Part 60.131 (a) requires suitable shielding to meet Title 10 CFR Part 20 occupational dose limits in repository
  - Title 10 CFR Part 60.135 (a) (2) requires radiolysis effects be considered
  - Title 10 CFR Part 60.135 (b) restricts materials allowable in the waste package, limiting shielding materials which can be used
    - » Non-corrosion enhancing
    - » Non-pyrophoric

Shielding Designs

- Waste Package
  - Shielding for radiolysis effects, which could cause corrosion
  - Use waste package barrier materials
- Waste Package Transporter and Other Facilities
  - Shield for personnel safety
  - Use multi-layer gamma and neutron shielding materials

Radiation Shielding Options

- Current design of waste packages provide sufficient shielding for protection of materials against radiolysis
  - Dose rates at surface approximately 16 R/hr
  - Radiolysis threshold approximately 1,000 R/hr
- Could shield waste packages for worker safety by
  - thicker metallic outer barriers (>390 mm) would improve containment
  - shielding sleeves, without containment credit, (>420 mm) create thermal problems
  - both are large and heavy

Radiation Shielding Options

(Continued)

- Approximate weights of shielding on transporter (gamma and neutron shielding)
  - Depleted uranium and boron-polyethelene: 39 tonnes
  - Lead and boron-polyethelene: 42 tonnes
  - Steel and boron-polyethelene: 54 tonnes

Radiation Shielding Options

(Continued)

- Transporter shielding more efficient and less costly
  - Not left in repository
  - Would required temporary shielding to enter emplacement drift
  - Would require radiation hardening any instrumentation in emplacement drifts

Future Radiation Shielding Activities

- Evaluate engineered barrier system and waste package component activation
- Evaluate radiation-induced corrosion products in accident conditions
- Consider multiple waste package interactions under radiation

Future Radiation Shielding Activities

(Continued)

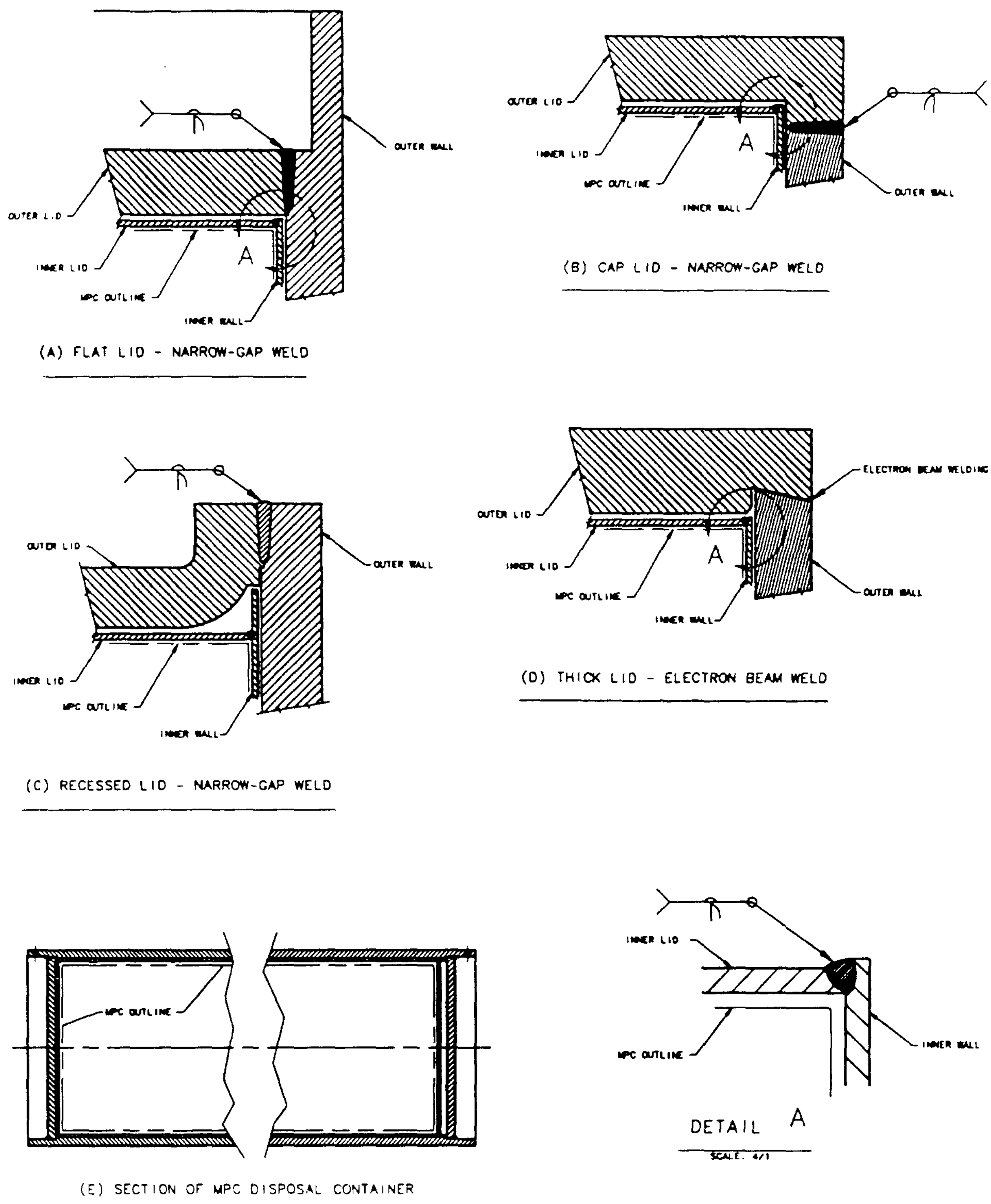
- Develop shielding needs for robotic/remote handling systems
- Evaluate emplacement drift shield door requirements
- Calculate emplacement operation radiation dose rates

Engineering Development Program Scope/Objectives

Develop methods and processes for waste package fabrication, remote closure, and inspection

Development Targets

- Fabrication techniques
  - Minimize stresses
  - Technically and economically acceptable
- Closure joint configurations
  - Narrow-gap welding conceptual design selection
  - Evaluate other remote welding techniques and methods



WASTE DISPOSAL CONTAINER CLOSURE JOINT CONCEPTUAL DESIGNS

Development Targets

(Continued)

- Closure Joint Configurations (Continued)
  - Establish joint design and weld process/ parameters
  - Minimize stresses
- Inspection of closure joint
  - Remote inspection methods required

Development Targets

(Continued)

- In-service inspection
  - Remote inspection methods required
  - Monitor waste package performance in repository
  - Required post-emplacement until repository closure

Development Targets
(Continued)

- Filler material
  - Criticality control
  - Chemical buffering for radionuclides
  - Cathodic protection
  - Function as mechanical packing
  - Improve thermal conductance

Time Table for Waste Package Engineering Development

DEVELOPMENT TARGETS	FY					
	95	96	97	98	99	00
FABRICATION TECHNIQUES		X	X	X		
CLOSURE JOINT CONFIGURATIONS	X	X	X	X		
INSPECTION OF CLOSURE JOINTS	X	X	X			
IN-SERVICE INSPECTION			X	X		
FILLER MATERIAL		X	X			

Linkage with Corrosion Research Program

- Edison Welding Institute interface
- Welded samples in material testing program
- Radiolysis catalized corrosion

Cooperation with Other Countries

• Focus '91 Conference	Sept. 1991
• Prague meeting	Sept. 1993
• Waste Package Workshop	Sept. 1993
• Kyoto meeting	Oct. 1994
• International High-Level Radioactive Waste Management Conference Papers	94/95
• BW Fuel Co/Cogema Relationship	On-going