HNF-2208-VA

Responses to Additional JAPC Questions

Prepared for the U.S. Department of Energy

Fluor Daniel Hanford, Inc.
Richland, Washington

Hanford Management and Integration Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200

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T. M. Burke
B&W Hanford

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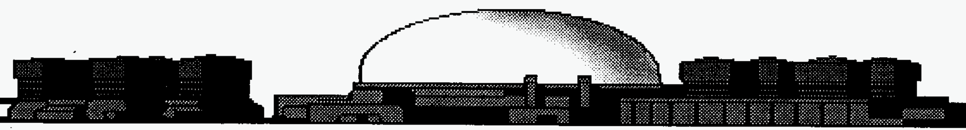
RESPONSES TO ADDITIONAL JAPC QUESTIONS

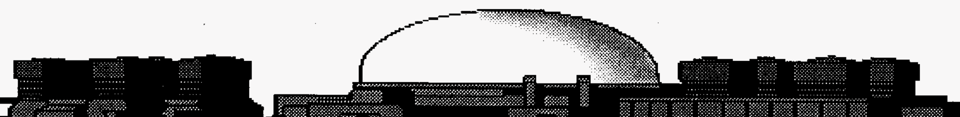
February 1998



POTENTIAL RELAXATION OF FUEL MANUFACTURING TOLERANCES

J. O. Dittmer

B&W Hanford Company



FUEL FABRICATION TOLERANCES/ STUDY FOCUS

The FFTF MOX fuel was designed and fabricated to very tight tolerances to reduce analytical uncertainties on design basis accident scenarios.

Is there sufficient data available to relax some of those tolerances and reduce fabrication costs, while not impacting the reliability or capability of the driver fuel to meet mission needs?

FIGURE 1
SCHEMATIC OF FFTF SERIES I AND II DFA FUEL PIN

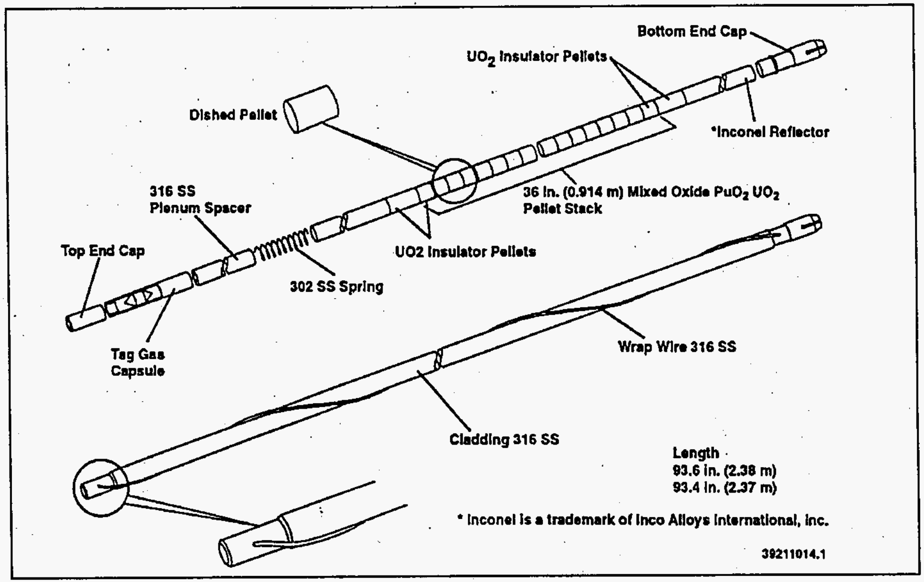
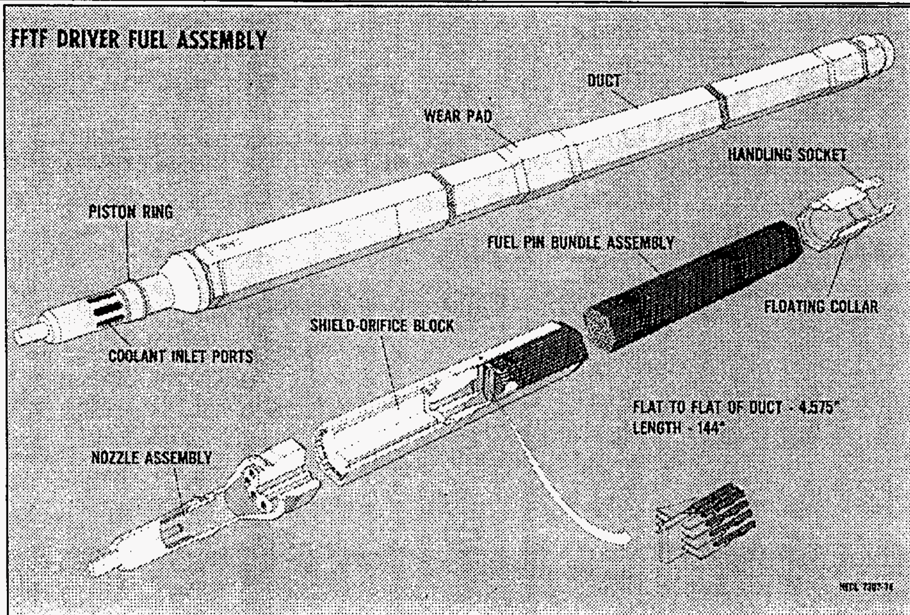


FIGURE 2
SCHEMATIC OF SERIES I AND II DRIVER FUEL ASSEMBLY



FFTF OPERATIONAL RESULTS SUMMARY: SERIES I & II DRIVER FUEL ASSEMBLIES (DFAs)

- Irradiated 210 DFAs
- Irradiated 119 test DFAs - included advanced designs
- Enrichment varied from 22.4 to 29.3% $\left(\frac{Pu}{Pu+U}\right)$
- DFA design goal of 80 MWd/kgM
- No fuel pin breaches to design goal peak burnup
- Provides bases for potential fabrication tolerance reduction

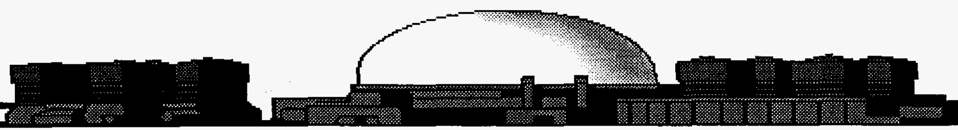


TABLE 1
KEY DESIGN CRITERIA* AND PERFORMANCE EVALUATION ITEMS

1	Cladding strain limits Plastic and thermal creep of cladding shall not exceed 0.2% at steady state. (This does not include irradiation creep)
2	Cladding wastage Fuel-cladding chemical interaction Sodium-cladding chemical interaction Fabrication scratches Fabrication dimensional tolerances Fretting and wear outer surface Contingency
3	Fuel/cladding mechanical interaction causing inelastic cladding strain
4	Bundle/duct interaction Axial Radial
5	Wear and fretting between wires and cladding
6	Pin cladding and wire differential
7	Fuel column axial stability (gaps between pellets)
8	Fuel melting and restructuring
9	Fuel fissile redistribution
10	Fission gas release from fuel
11	Fuel burnup
12	Irradiation swelling of cladding and ducts

*RDT standards used for fabrication.

SUMMARY OF IRRADIATION TESTS ON DFA AND FUEL FABRICATION VARIABLES

- Goal: Improve performance and reduce costs
- Variables tested (tables 2 and 3): Fuel fab and assembly tolerances and cladding materials
- Significant results:
  - Goal lifetimes achieved
  - D9/HT9 alloys superior: Reduced swelling potential
  - Duct mechanical attachment methods viable
  - Test performance per design predictions (Figure 3)

TABLE 2
FUEL FABRICATION VARIABLES AND NUMBER OF FUEL ASSEMBLIES TESTED

FUEL VARIABLES	NUMBER OF TEST ASSEMBLIES*
Pu content [Pu/(Pu + U)]	14
Flat end pellets	22
Annular pellets	15
Pellet density	22
Smeared density	23
O/M	15
Pellet diameter	24
Fuel-cladding gap	23
Gel sphere feed	1
Fuel pin diameter	21
Pellet fabrication defects	1
Weight-to-length	12
Cladding thickness/diameter	21

*Individual test could have and usually did have multiple fuel fabrication variables tested.

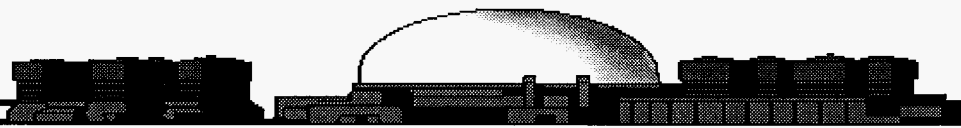
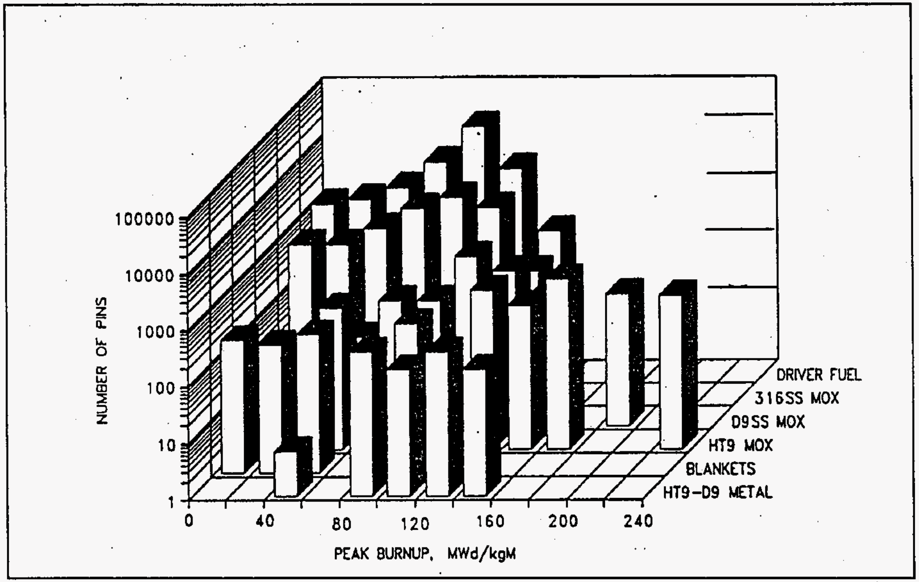


TABLE 3
ASSEMBLY VARIABLES AND NUMBER OF FUEL ASSEMBLIES TESTED

ASSEMBLY VARIABLES	NUMBER OF TEST ASSEMBLIES*
Cladding composition	26
Pin-bundle spacing	8
Bundle-to-duct clearance	6
Wire wrap pitch	23
Duct alloy	31
Duct attachment	16
End cap weld	14

*Individual test could have and usually did have multiple assembly variables tested.

FIGURE 3
STATUS OF BURNUP IN DFA AND TEST PINS IN FFTF



CDE INVESTIGATED RELAXATION OF FUEL FAB SPECIFICATIONS

- CDE fab specs based on updated RDT standards and technical requirements
- Major tolerance differences from Series II DFAs (design values and pellets)
- CDE utilized larger fuel pin diameter
- Test results indicate:
  - Successful fuel pin exposure to 230 MWd/kgM
  - Reduced fab costs through fewer operational steps and rejects
  - HT9 alloy superior to austenitic steels

EFFECTS OF INCREASED PIN DIAMETER ON FUEL PERFORMANCE

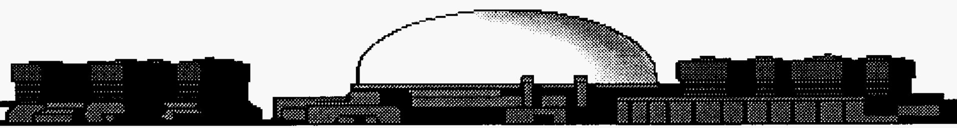
- Irradiation tested larger diameter pin in 169 pin assembly
  - 0.275-inch diameter pin with 0.015-inch wall thickness
  - 0.270-inch diameter pin with 0.022-inch wall thickness
- All configurations achieved 99.9% reliability goal
- Achieved increased burnup and fluence with 0.022-inch wall thickness
- Cladding thickness-to-diameter ratio is a critical parameter

FUEL FABRICATION RELAXATION CANDIDATES

- Pellet end configuration:
  - Flat ends: Large potential savings, need confirming analysis for transient events
  - Dished ends: Potential savings with use of small samples for inspection
- O/M limits: Reduce lower pin limit from 1.94 to 1.93 needs confirming analysis for transient events
- Cracks, chips, voids: Good basis for relaxation. Corner chips need more analysis

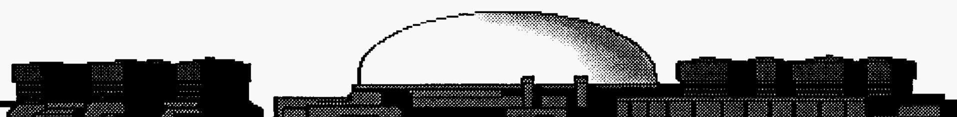
RECOMMENDATIONS FOR POTENTIAL FUTURE FUEL FABRICATION SAVINGS

- Use flat ended pellets
- Reduce lower o/m limit to 1.93
- Adopt sinter-to-size pellets
- Utilize programmed startup
- Conduct a "risk vs. cost" benefit analysis
- Develop regulatory acceptance strategy using analysis techniques



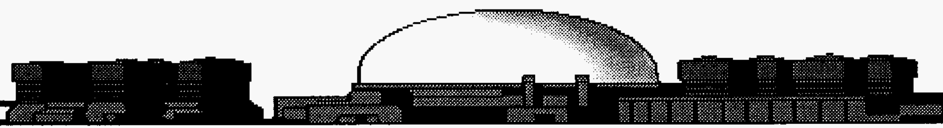
DESCRIPTION OF AN ALTERNATE CORE DESIGN FOR ENHANCED SAFETY

David Wootan
Fluor Daniel Northwest, Inc.



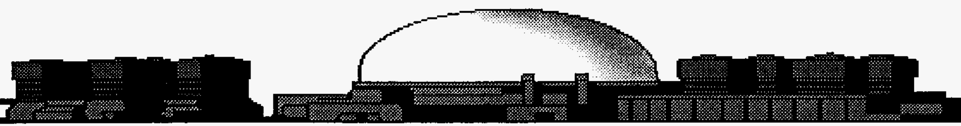
HANFORD EXPERIENCE

- FFTF operation and safety analysis
- Variety of fuel types, core configurations
- Passive safety tests
- GEMS were developed and tested to mitigate loss of flow accidents
- What are some other options for enhancing reactor safety?



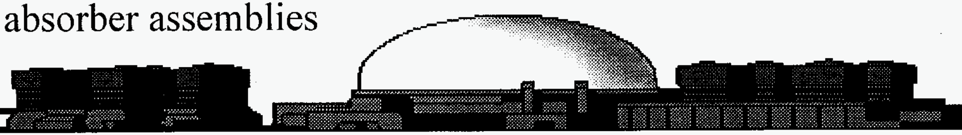
AN ALTERNATIVE CONCEPT FOR ENHANCING SAFETY

- Adding significant number of absorber assemblies
  - Increases fuel enrichment
  - Hardens neutron spectrum
  - Reduces Doppler reactivity feedback
  - Reduces positive coolant void reactivity feedback
- Potential beneficial impact on unprotected transients
- No apparent penalties on protected transients

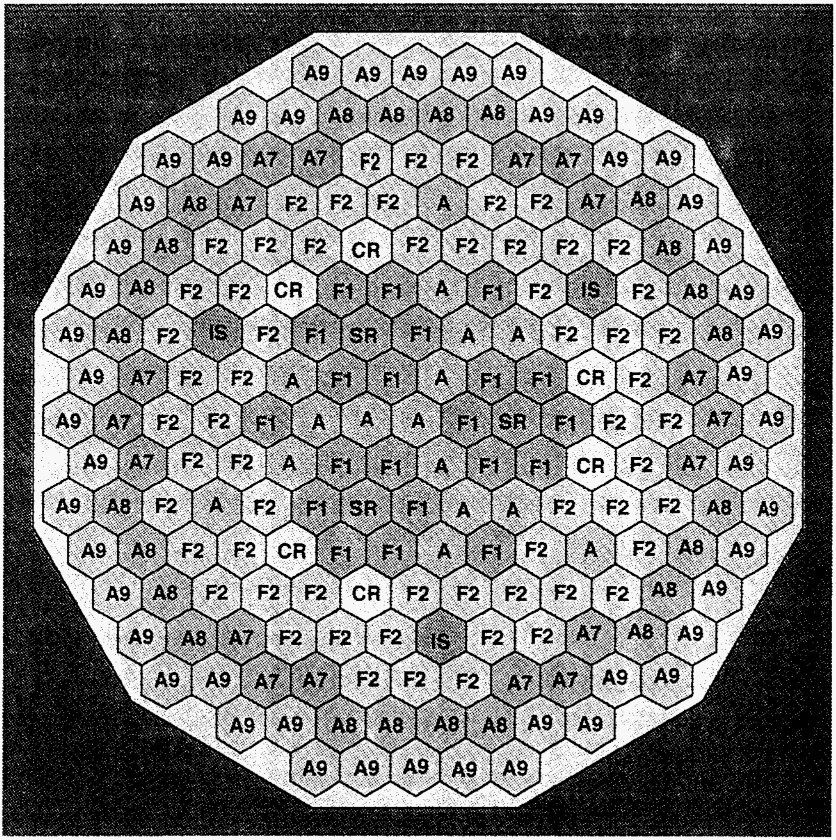


FFTF METHODS AND DATA USED TO EVALUATE SAFETY PERFORMANCE

- FFTF core model
  - Methods verified by operating data
  - FSAR available for comparison
- Reference core configuration
  - Heterogeneous core arrangement
  - No axial or radial blankets
  - No radial reflectors
  - Pu enrichment $\leq 40\%$ experience base
  - Boron carbide absorber assemblies
    - 16 in-core, 90 peripheral
    - 61 pin control rod fixed shim design
    - 125 cm active length
- Alternative core configurations with six and zero in-core absorber assemblies



Reactor Core Map For Enhanced Safety Core



A = Fixed Absorber In Core
A7 = Fixed Absorber Row 7
A8 = Fixed Absorber Row 8A
A9 = Fixed Absorber Row 8B&9
IS = In Core Shim

F1 = Inner Zone Fuel (Row 1-4)
F2 = Outer Zone Fuel (Row 5-7)
CR = Control Rod (Secondary)
SR = Safety Rod (Primary)

VALIDATED METHODS

- FFTF core reload design procedures
- Operating parameters compared to technical specification limits
- 10 years FFTF operating experience
- FSAR-quality analyses
- 2D, 3D diffusion theory
- First order perturbation theory
- Selected Monte Carlo used for confirmation

KEY SAFETY PARAMETERS

- Fuel enrichment
- Fuel peak linear power
- Fuel and absorber worth
- Control and safety rod worths
- Radial expansion coefficient
- Axial expansion coefficient
- Doppler constant
- Sodium void reactivity
- Reactivity worth distributions

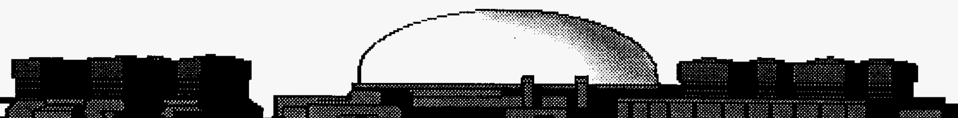
COMPARISON OF BORON CARBIDE AND FSAR CORE PARAMETERS

PARAMETER	UNITS	FSAR CORE	Reference B_4C Core	
			BOEC	EOEC
Doppler constant	(Tdk/dT)	-0.005	-0.00055	-0.00069
Uniform Axial Expansion	(dk/k per cm)	-0.003	-0.0055	-0.0050
Uniform Radial Expansion	(dk/k per cm)	-0.0125	-0.0090	-0.0093
Total Sodium Void	(dk/k)	-0.013	-0.0065	-0.0082
Delayed Neutron Fraction		0.00318	0.00276	0.00279

- Reactivity feedbacks compared to FSAR core
  - Slightly lower radial expansion feedback
  - Factor of 1.8 higher axial expansion coefficient
  - Factor of 10 lower Doppler constant
  - Total fuel sodium void is less negative

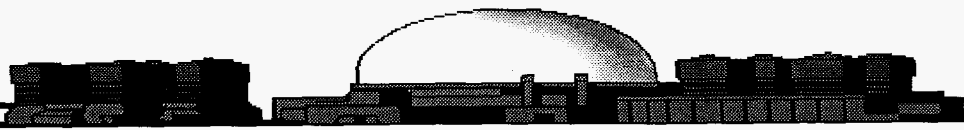
VARIATION OF SAFETY-RELATED PARAMETERS WITH NUMBER OF IN-CORE FIXED ABSORBERS

PARAMETER	16 IN-CORE ABSORBERS	6 IN-CORE ABSORBERS	0 IN-CORE ABSORBERS
Doppler constant (Tdk/dT)	-0.00055	-0.00130	-0.00265
Core radial expansion coefficient (dk/k per cm)	-0.0090	-0.0089	-00091
Fuel axial expansion coefficient (dk/k per cm)	-0.0055	-0.0049	-0.0042
Worth of voiding sodium from core region of fuel assemblies (dk/k)	-0.0042	-0.0030	-0.0020
Positive sodium void region in fuel assemblies (dk/k)	0.0051	0.0073	0.0081
Maximum positive void region in one assembly (dk/k)	0.00016 (in 2303)	0.00039 (in 2101)	0.0039 (in 2101)



AS NUMBER OF IN-CORE ABSORBERS IS REDUCED:

- Fuel enrichment reduced
- Neutron spectrum softened
- Doppler coefficient becomes more negative
- Radial and axial expansion coefficient insensitive
- Maximum positive sodium void coefficient in a single assembly increased
- Magnitude of positive sodium void region increased

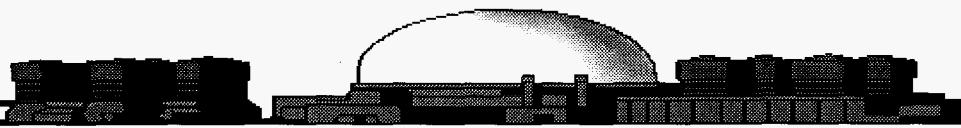


REACTIVITY EFFECTS OF SODIUM VOIDING OF REFERENCE CORE

REGION VOIDED	REACTIVITY EFFECT, dk/k ($)	
	BOEC	EOEC
All fuel assemblies over active fuel height (92.28 cm)	-0.0065 (-2.4$)	-0.0082 (-2.9$)
Central positive void region of the 3 Row 2 fuel assemblies	0.00038 (0.14$)	0.00037 (0.13$)
Maximum positive void region in fuel assembly	0.00016 (0.058$)	0.00014 (0.050$)
Positive void region in all fuel assemblies	0.0051 (1.8$)	0.0046 (1.6$)

CONCLUSIONS

- Identified an alternative concept for improving safety with potential application to DFBR
- Adding fixed absorber assemblies
  - Provides more attractive reactivity feedbacks
  - All operational limits satisfied, including component worths, shutdown margins, cycle length, peak linear power, temperatures
- Other potential applications
  - Increased capacity factor by varying absorber loading
  - Alternate absorbers to produce beneficial isotopes



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

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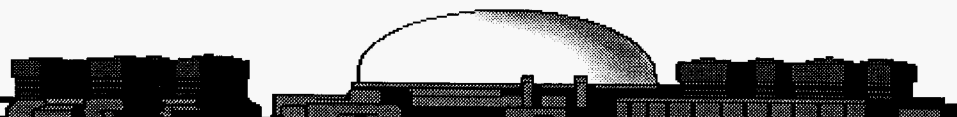
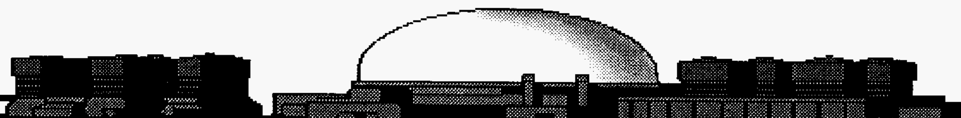


SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

DISCUSSION TOPICS

- Purpose
- Model development
- Analysis methodology
- Results
- Summary and conclusions



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

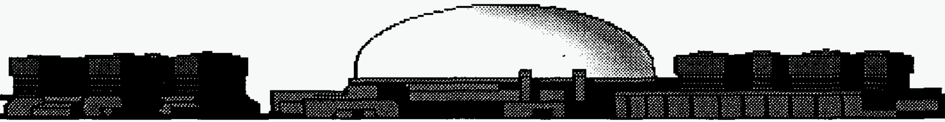
PURPOSE

- Perform safety assessment of alternate core designs
  - Unprotected transients (ULOF and UTOP)
  - 3 variations in absorber loading
  - Significant changes in reactivity and neutronic parameters
- Evaluate sensitivity of alternate core designs
  - Fuel Doppler temperature coefficient
  - Fuel axial expansion
  - Core radial expansion coefficient
- Evaluate impact on core margins
  - Margin-to-Sodium boiling (ULOF)
  - Extent of fuel melting (UTOP)

SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

MODEL DEVELOPMENT

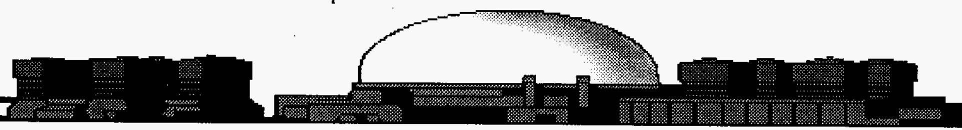
- SAS4A/SASSYS-1 Transient Model
- Core model
  - Reference (16 in-core) fixed absorber assemblies
  - 41 channels; 31-fuel, 10-absorber
  - Approximated alternate designs via adjustable reactivity coefficients
  - Separate thermal pin models for driver fuel and absorbers
- Balance-of-Plant model
  - Three FFTF loops represented by two SASSYS loops
  - Reactor control systems
  - Reactor shutdown systems



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

MODEL DEVELOPMENT (cont)

- Required input (data)
  - Mechanical design
  - Neutronic
  - Thermal-hydraulic
  - Thermo-physical material properties
  - Balance-of-plant
- Axial expansion reactivity feedback models
  - Simple
    - Used with fixed absorber assemblies
    - Based on differential thermal expansion
    - Cladding controlled force balance
  - Detailed
    - Used with driver fuel assemblies
    - Based on fuel-clad interaction (FCI) performance analyses
    - Cladding controlled axial-plane strain model for ULOF
    - Mixed axial-plane strain model for UTOP



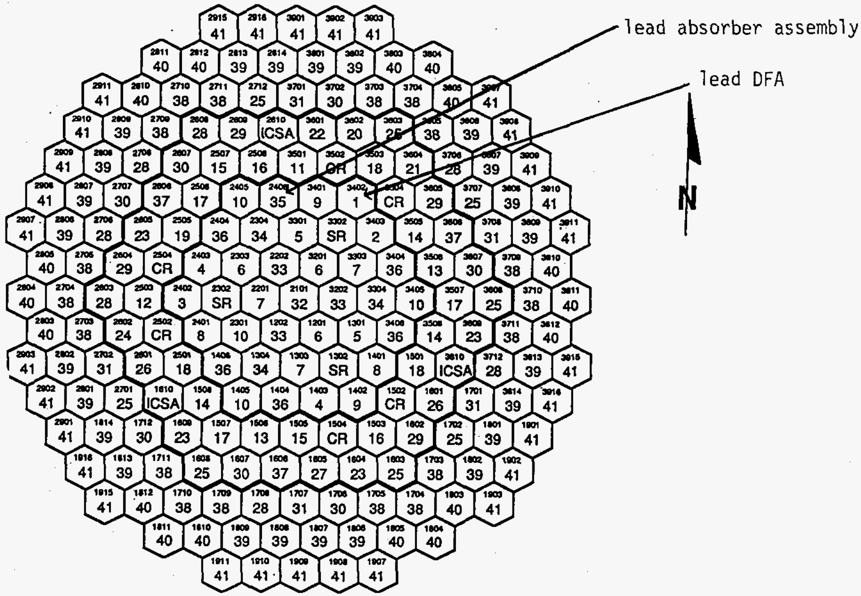
SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

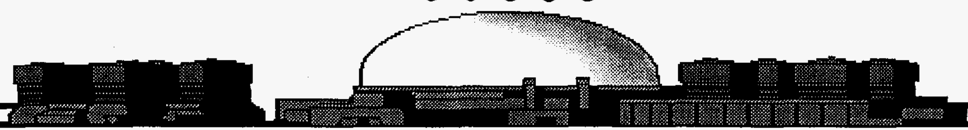
MODEL DEVELOPMENT (cont)

- Core radial expansion reactivity feedback model
  - Detailed mechanistic beam model
  - Three point contact and restraint treatment
  - Accounts for assembly bowing
  - Validated and verified against FFTF data
  - Based on full compliment of DFA in Rows 1-6
  - Increased uncertainty when applied to alternate core designs
- Initiating events and programmed system features
  - —ULOF; loss-of-forced-flow (primary pump torque = 0.0)
  - —UTOP; reactivity insertion (0.50$/s, 4$ total)

SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

Core Location-to-Channel Assessments



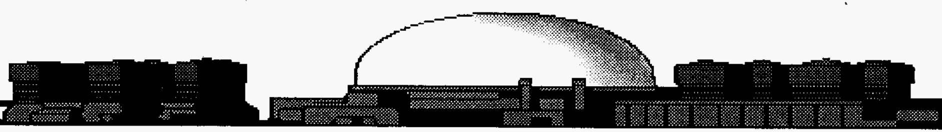


SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

ANALYSIS METHODOLOGY

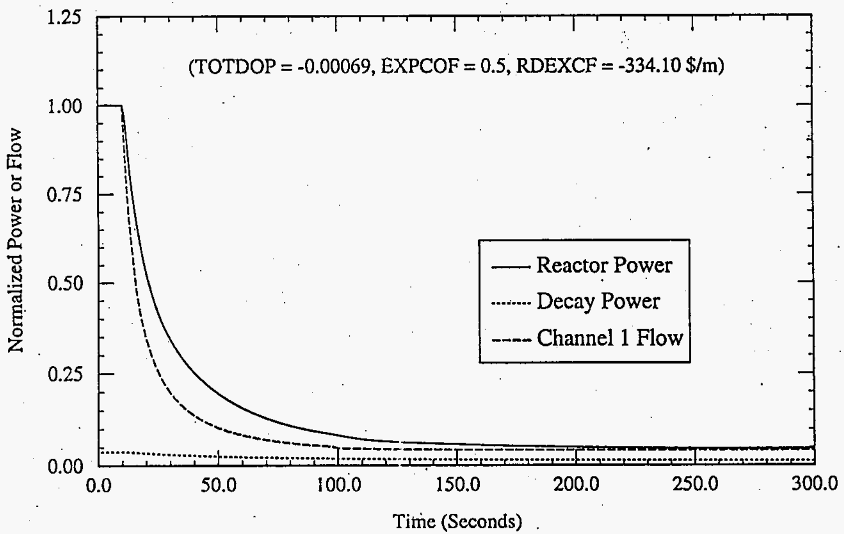
Series of parametric analyses

- Reference (16 in-core) fixed absorber loading
- Corresponding to EOEC conditions
- Total fuel Doppler coefficient (TOTDOP)
  - 16 in-core absorbers, -0.00069
  - 6 in-core absorbers, -0.00130
  - 0 in-core absorbers, -0.00265
  - Other values, -0.004, -0.005, -0.006
- Fuel axial expansion multiplier (EXPCOF); 1.0, 0.5, 0.0
- Core radial expansion coefficient ($/m) (RDEXCF); -334.10, -167.05
- Base case (TOTDOP = -0.00069, EXPCOF = 0.5, RDEXCF = -334.10)
- Reference condition: 400MW, 633°K (680°F) inlet temperature, 143°K (258°F) reactor vessel temperature rise



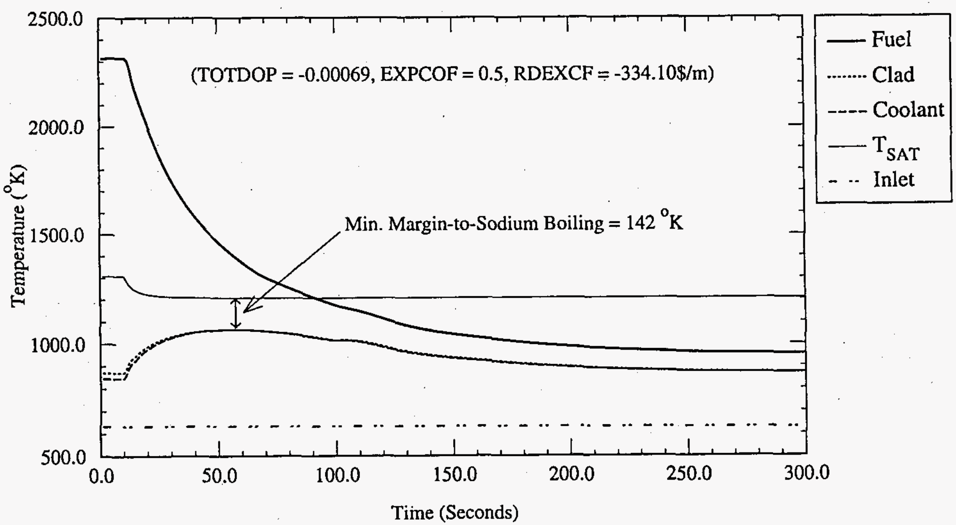
SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

Normalized Reactor Power, Decay Power, and Channel 1 Flow Versus Time
During a ULOF for the Reference Fixed Absorber Loading



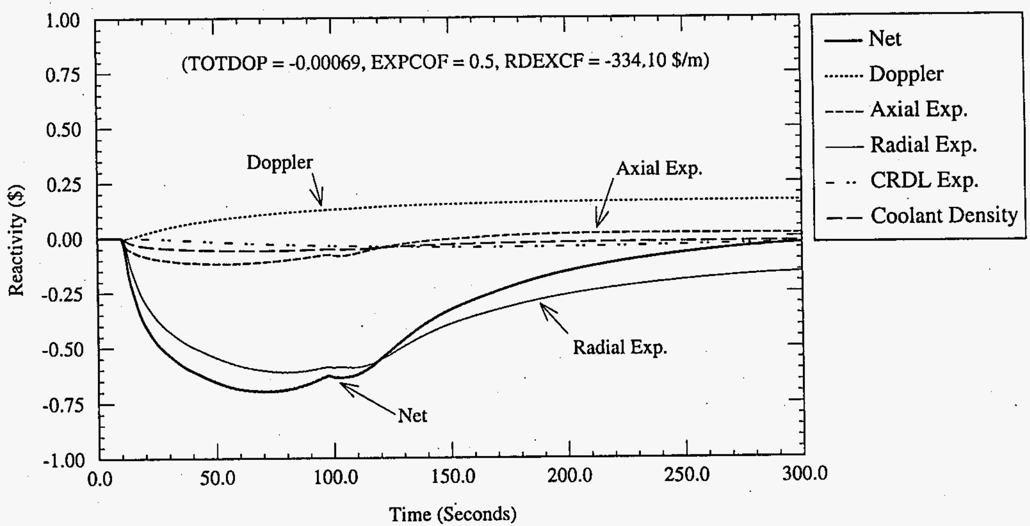
SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

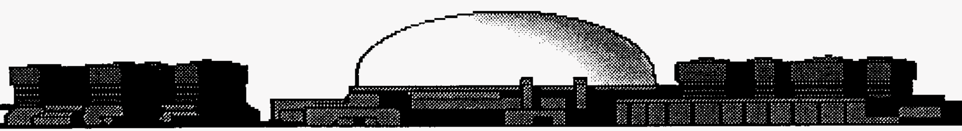
Peak Fuel, Clad, Coolant, and Inlet Temperatures for Channel 1 Versus Time During a ULOF for the Reference Fixed Absorber Loading



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

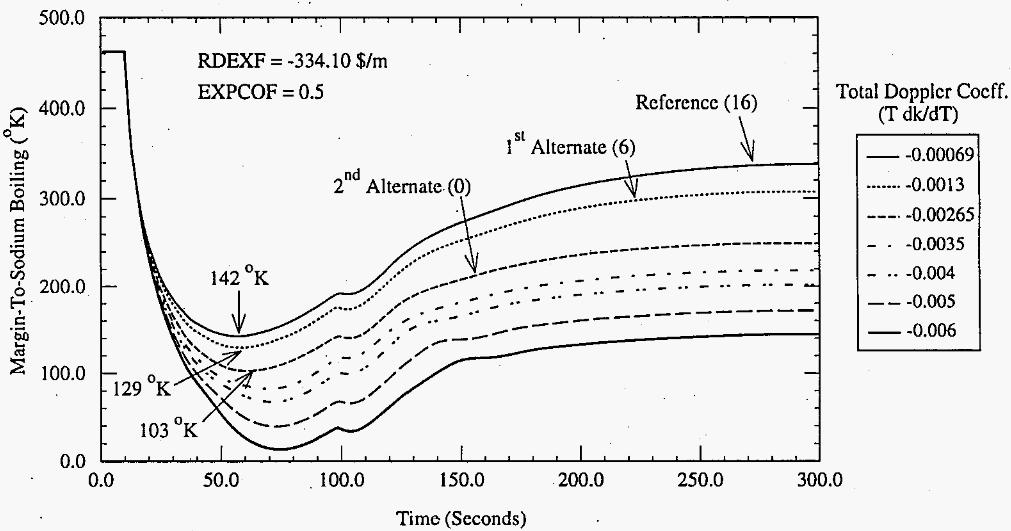
Components to Net Reactivity Versus Time During a ULOF for the Reference Fixed Absorber Loading





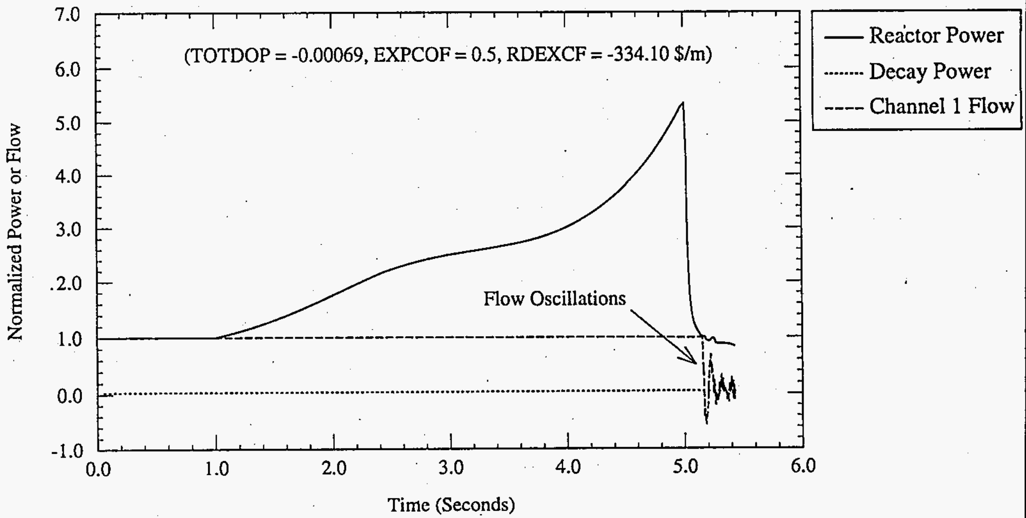
SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

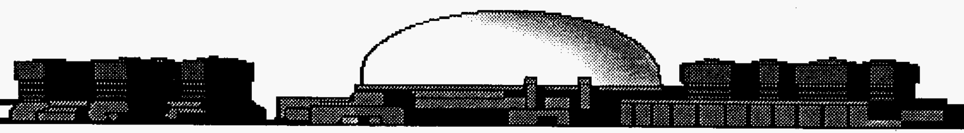
Margin-To-Sodium Boiling Versus Time During a ULOF for Variations in the Total Doppler Coefficient for the Reference Fixed Absorber Loading



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

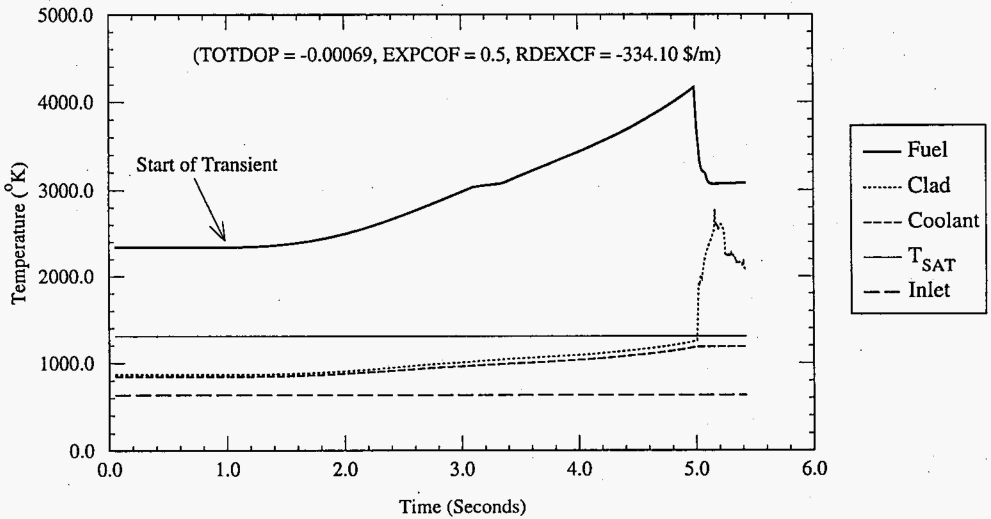
Normalized Reactor Power, Decay Power, and Channel 1 Flow Versus Time
During a UTOP for the Reference Fixed Absorber Loading





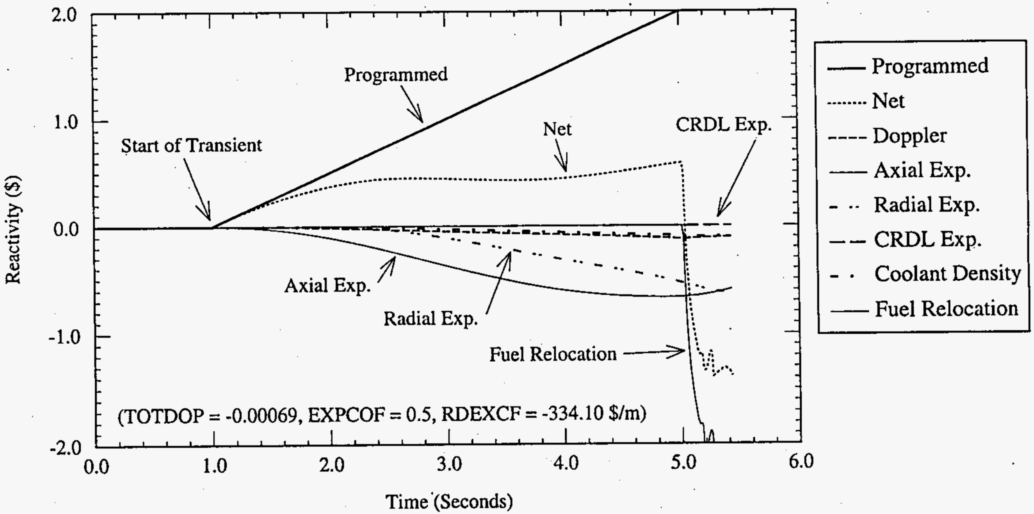
SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

Peak Fuel, Clad, Coolant, and Inlet Temperatures for Channel 1 Versus Time
During a UTOP for the Reference Fixed Absorber Loading



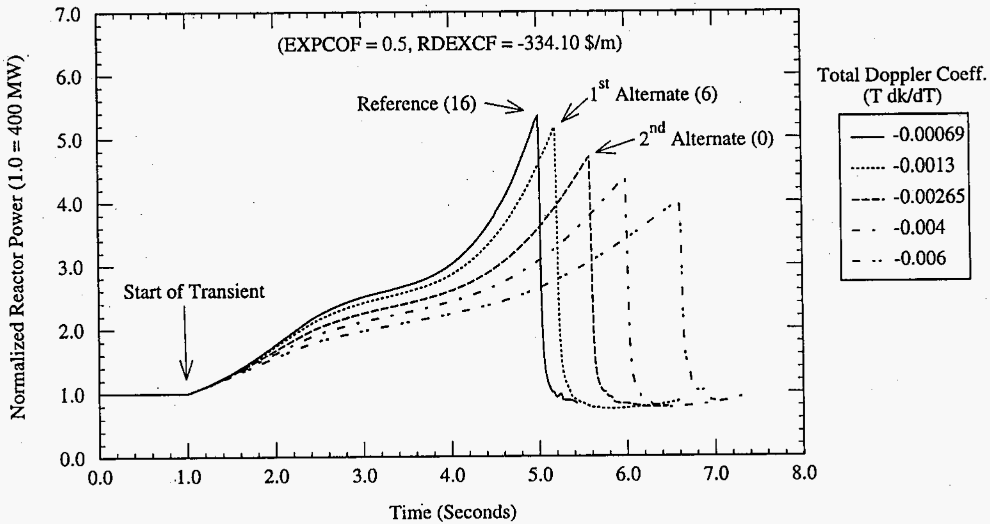
SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

Components to Net Reactivity Versus Time During a UTOP for the
Reference Fixed Absorber Loading



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

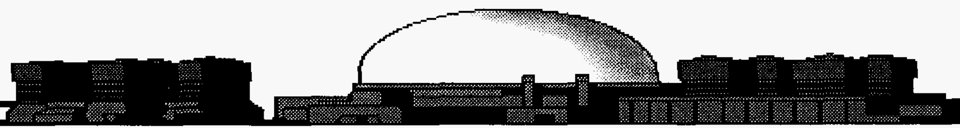
Normalized Reactor Power Versus Time During a UTOP for Variations in the Total Doppler Temperature Coefficient for Reference Fixed Absorber Loading



SAFETY ASSESSMENT OF UNPROTECTED TRANSIENTS FOR ALTERNATE CORE DESIGNS

SUMMARY AND CONCLUSIONS

- ULOF
  - Safety margins have increased (no sodium boiling)
  - Core radial expansion is dominant negative reactivity feedback
  - Greater uncertainty with core radial expansion due to "mixed" inner core loading
- UTOP
  - Safety margins unchanged (results similar to FFTF FSAR)
  - Fuel relocation required to terminate transient
  - Axial expansion is dominant negative reactivity feedback prior to fuel relocation
- **Safety margins for ULOF have increased without a detrimental impact on UTOP**



AREAS OF POSSIBLE FUTURE COLLABORATION

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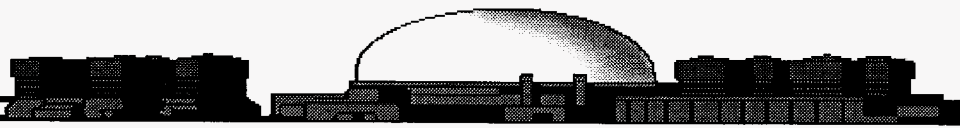
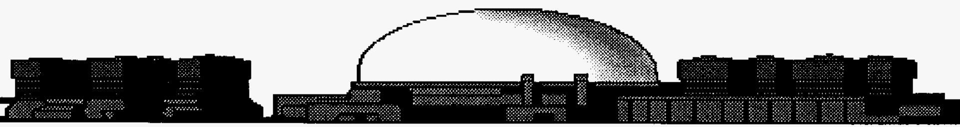


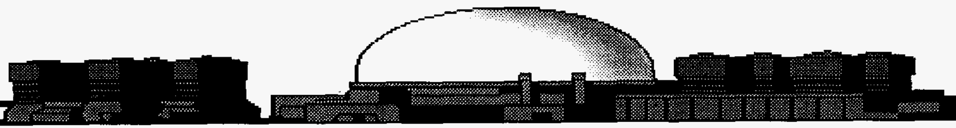
CONTINUE SUPPORT OF GEM/DFBR EVALUATIONS

- Review JAPC GEM/DFBR design and analyses
- Advise JAPC on required analyses and testing
- Respond to additional JAPC questions
- Assess structural feedback models by analyzing GEM pump start tests with SASSYS



CONTINUE WORK ON ALTERNATE CORE DESIGNS

- Variations in core configuration
  - Number and location of absorbers
- Effect of core size
- Fuel types
- Absorber materials
- Further understanding/enhancement of feedbacks
  - Axial fuel expansion
  - Radial expansion and bowing
- Application of concepts to actual core designs and development of operational strategies



AREAS OF POSSIBLE COMMON INTEREST

- Sodium spill and fire
  - Component (e.g., thermowell) design and testing
  - Sodium leak accommodation
    - Leak detection
    - Spill containment
    - Fire suppression (e.g., nitrogen flooding)
  - Sodium fire analyses and testing
- Fuel Manufacturing
  - Manufacturing processes/equipment
  - Fuel design/manufacturing tolerances
  - Facility fire accommodation (e.g., ventilation system design)
- Medical isotope production

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B. Document Number HNF-2208 (VA)

C. Title Responses to Additional JAPC Questions

D. Internet Address Advanced Reactors Transition

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