MASTER

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M.I.T. SUMMER COURSE

FAST REACTOR SAFETY

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SAFETY ASPECTS OF FAST REACTOR FUELS

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FUELS SAFETY

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LMFBR SAFETY TECHNOLOGY

LINES OF ASSURANCE

- 1 -- PREVENT CDA'S
- 2 -- LIMIT CORE DAMAGE
- 3 -- CONTROL CDA PROGRESSION
- 4 -- ATTENUATE RADIOLOGICAL CONSEQUENCES

LMFBR SAFETY TECHNOLOGY

PROGRAM STRUCTURE

LOA - 2 ISSUES AND TECHNOLOGY

ISSUES

TRANSIENT OVERPOWER

- REACTIVITY RATE
- INTERNAL AXIAL FUEL MOBILITY
- FAILURE TIME, LOCATION
- FUEL MOBILITY
- FUEL RELOCATION
- SUSTAINED SHUT-DOWN COOLING

LOSS OF COOLING

- LOSS OF PUMPING, PIPING INTEGRITY OR BLOCKAGE FORMATION
- FUEL/CLADDING AXIAL AND RADIAL THERMAL EXPANSION
- THERMAL HYDRAULICS
- SODIUM VOID AND CLADDING REACTIVITY WORTH
- FUEL PIN FAILURE TIME AND LOCATION
- FUEL CLADDING AND FISSION GAS DYNAMICS
- ENERGETICS
- FUEL RELOCATION
- SUSTAINED SHUT-DOWN COOLING

TECHNOLOGY AREAS

- FUEL CLADDING, FISSION GAS, SODIUM PROPERTIES
- SUBASSEMBLY THERMAL HYDRAULICS
- FUEL PIN FAILURE MECHANISMS
- FUEL PIN MATERIAL DYNAMICS
- ENERGETICS
- MECHANICAL CONSEQUENCES
- SUSTAINED SHUT-DOWN COOLING

COMPARISON OF LWR AND LMFBR FUELS

PARAMETER	FTR/CRBR	LWR
FUEL	90% DENSE (Pu, U)O ₂	93% DENSE UO ₂
POWER DENSITY	220 W/gm	50 W/g
FUEL PELLET DIAMETER	0.194 IN.	0.488 IN.
COOLANT INLET TEMPERATURE	600-800°F	380°F
LINEAR HEAT RATING	12.7/12.6 kW/ft	18.3 kW/ft
DESIGN MELTING LINEAR HEAT RATING	16-19 kW/ft	23 kW/ft
DESIGN OVERPOWER HEAT RATING	15.3/14.4 kW/ft	21 kW/ft
FUEL MELTING TEMPERATURE	5000°F	5180°F
MAXIMUM HEAT FLUX	867,000 B/hr-ft ²	425,000 B/hr-ft ²
MAXIMUM BURNUP	80 MWd/kg	40-50 MWd/kg

HEDL 7605-109.16

FFTF DRIVER FUEL PIN

STRUCTURE EVOLUTION IN AN OXIDE FUEL PIN

NOMINAL HEAT RATING ≈ 13 kW/FT THROUGHOUT LIFE

WHAN-PR-4

J. A. Christensen
WADCO Corporation
Richland, Washington
for the U.S.A.E.C.
under contract no. AT(45-1)-2170

- ① As H but with Family of New Cracks Formed on Shutdown
- ② Retains large voids formed Early in Life in Columnar Grains
- ③ Pore Size Refined Beyond Columnar Grains (Fission Gas Bubbles)
- ④ No Gap Remaining for Large H_2 (~ 2000 SLU/HR, 11 F)
- ▲ Voids Between 75 and 1000 Å Depending on Clad Temperature; Void Density Increases with Burnup; Void Size Increases with Temperature; Uniform Void Distribution with Slight Grain Boundary Deviation
- ▲ Dislocation Loops Commensurate with Atoms Displaced from Void Sites
- ▲ Possible Recrystallization with 900 F Na Outlet Temperature
- ▲ Grain Boundary σ Phase Concentration Saturated at 5-7%
- ⑤ Heat Rating Reduced by Fissile Depletion
- ⑥ Largest Holes and Old Cracks Replaced by Fine ($< 1 \mu$) Gas Bubbles and Inclusions
- ⑦ Extensive Fission Product and Clad Derived Inclusions
- ⑧ Fission Product Oxides in Equiaxed Grains; Fission Product Metals in Columnar Grains Concentrated Near Edge
- ▲ Localized Cladding Strain Near Cracks
- ⑨ Cladding Swelling Severe ($\sim 10\%$, Max Cause Gap to Reopen?), Maximum Pin Diameter
- ▲ Ductility Decreased to Zero
- ⑩ Maximum Impurity and Irradiation Effect on Thermal Conductivity and Melting Point
- ▲ Severe Localized Internal Grain Boundary Oxidation When I.D. Temperatures $> 1000^\circ\text{F}$

During Operation
After Shutdown

- ① Fuel-Cladding Gap
- ② Pores, Voids, Cracks
- ③ Metallic Fission Fragments
- ④ Non-Metallic Fission Fragments
- ⑤ Cladding
- ⑥ Structural and Dimensional Features
- ⑦ Temperature Distribution Effects
- ▲ Effects on Clad (10 MW/MTM ~ 10 nvt $> 1\text{MeV}$)

- ① (U, Pu)O₂ $\times$ 10/Pu ~ 3 ; O/M ~ 1.53
- ② Small ($\sim 10^{-3}$) Equiaxed Grains
- ③ Equilibrium Pores to $\sim 50\%$
- ④ Large Voids, $\sim 70\%$
- ⑤ Small Cracks, Cracks, etc.
- ⑥ Gap Large (~ 3 mil) and Uniform, H_2 Filled
- ▲ 316 Stainless Steel Cladding

- ① Radial and Circumferential Cracks from Thermal Shock
- ② Startup Cracks Taper Inward
- ③ Gap Partly and Irregularly Closed from Thermal Expansion and Cracking; He Contaminated by Desorbed Gases.
- ④ Reduction in Thermal Conductivity Because of Cracks

- ① Columnar Grain Formation by Lenticular Pore Migration
- ② Central Void Formed from Pores Displaced from Columnar Grain Region
- ③ Some Equiaxed Grain Growth, Pores Move to Grain Boundaries
- ④ Large Stable Central Pores — Increase in Size and Density Toward Center
- ⑤ Small Migrating Spherical Pores
- ⑥ Crack & Hole Healing to Beyond Grain Growth Region
- ⑦ Gap Unchanged from "B" (or Perhaps Enlarged by Sintering or Closed by "Bloating")
- ▲ Grain Boundary Carbide Precipitate Concentration Reaches Limiting Value with 1100 F Na Outlet Temperature

- ① New Family of Cracks Which Taper Outward Formed During Shutdown
- ② Gap Opens Some Compared with "C" but Not to Condition in "A" Because Thermal Expansion Strains (cracks) Remain and are Partially Redistributed Toward Center
- ③ Thermal Conductivity Increased Because of Porosity Depletion in Higher Temperature Regions
- ▲ Observable Grain Boundary σ Phase with 1100 F Na Outlet Temperature

- ① Gap Closes Further
- ② New Cracks Heal — Associated Void Redistributes Toward Center
- ③ Gap Conductance Decreases from Xe and Desorbed Gas Contamination
- ④ "Stable" Residual Crack Pattern Achieved After several Cycles
- ⑤ Grain Growth Proceeds; Columnar Grain Growth "Temperature" Decreases Continuously During Life
- ⑥ Central Void Diameter Increases Continuously During Life

- ① Bubble Steady State Achieved in Warmer Portions
- ② Large Voids and Old Cracks Begin to Close from Swelling
- ③ Fission Fragments and Clad Derived Inclusions Appear
- ④ Thermal Conductivity Further Decreased
- ▲ Internal Grain Boundary Oxidation for O/M Near 2.00 When I.D. Temperatures $> 1000^\circ\text{F}$

- ① Gap Conductance Reaches Maximum, Steady-State Value
- ▲ 1% Swelling from Voids; H_2 at Grain Boundaries Reduces Ductility to $\sim 1/2$ Pre-Irradiation Value
- ▲ Grain Boundary Carbide Precipitation with 900 F Na Outlet Temperature Reaches Limiting Concentration
- ▲ Recrystallization with 1100 F Na Outlet Temperature
- ▲ Observable Grain Boundary σ Phase with 900 F Na Outlet Temperature

- ① Gap Closed by Crack-Ratcheting After Several Cycles or Accelerated Fission Gas Swelling at Intermediate Temperatures (1300-1500 C)
- ▲ Clad Strain from Mechanical Interaction with Fuel on Cycling
- ② Residual Cracks Narrower Because of Clad Restraint and Swelling Accommodation
- ③ Slight Irradiation Decrease in Thermal Conductivity
- ▲ Ductility Decreased Because of H_2 in Grain Boundaries (~ 1 PPM H_2 , 10 nvt)
- ▲ Degraded Mechanical Properties Because of Decarburization by Sodium

THERMAL PERFORMANCE

DESIGN CRITERION

NO FUEL MELTING AT OVERPOWER
TRIP CONDITIONS

HEDL 7605-109.1

THERMAL PERFORMANCE

$$T_c = \Delta T_{\text{FUEL}} + \Delta T_{\text{GAP}} + \Delta T_{\text{CLADDING}} + T_{\text{COOLANT}}$$

WHERE

T_c = FUEL CENTERLINE TEMPERATURE

ΔT_{FUEL} = TEMPERATURE GRADIENT ACROSS FUEL

ΔT_{GAP} = TEMPERATURE GRADIENT ACROSS FUEL
CLADDING GAP

$\Delta T_{\text{CLADDING}}$ = TEMPERATURE GRADIENT ACROSS CLADDING

T_{COOLANT} = LOCAL TEMPERATURE OF COOLANT

THERMAL PERFORMANCE

$$\Delta T_{\text{FUEL}} = \int_{T_s}^{T_r} k dT = \frac{Q'}{4\pi} \left[\frac{a^2 - r^2}{a^2 - b^2} - \left(\frac{2b^2}{a^2 - b^2} \right) \ln \frac{a}{r} \right]$$

WHERE Q' = LINEAR HEAT GENERATION RATE, w/cm

a = FUEL PELLET OUTER RADIUS

b = FUEL PELLET INNER RADIUS

r = RADIUS OF INTEREST

$\int KdT$ FOR MIXED OXIDE FUEL

HEDL 7605-109.25

MARGIN TO MELTING ASSESSMENT

HEDL 7605-109.2

INFLUENCE OF INITIAL FUEL-CLADDING GAP SIZE ON THE LINEAR-HEAT-RATING-FOR- INCIPIENT-FUEL-MELTING IN MIXED OXIDE FUEL PINS.

FUEL-CLADDING POSTIRRADIATION GAP MODEL

HEDL 7602-113.5

INFLUENCE OF BURNUP ON HEAT-RATING-TO-MELTING FOR MIXED OXIDE FUEL

GRAIN GROWTH MODELS

EQUIAXED GRAIN GROWTH

$$T_E = C$$

HEDL 7510-173.1

COLUMNAR GRAIN GROWTH

$$T_C = A + B \left(\frac{\partial T}{\partial R} \right)$$

HEDL 7510-173.2

761596-

HEDL 7602-113.6

FISSION GAS RETENTION IN MIXED OXIDE FUEL

SAMPLE

B	C	D	E	F	G	H	I	J	K	L	M	N
---	---	---	---	---	---	---	---	---	---	---	---	---

0.0625"

6.75"

13.5"

HEDL 7410-99.5

RETAINED FISSION GAS DISTRIBUTION HIGH POWER PIN

7112-25.14

DIMENSIONAL STABILITY OF FUEL

BURNUP DEPENDENCE OF FUEL COLUMN LENGTH CHANGE FOR EBR-II IRRADIATED MIXED OXIDE FUEL PINS

CUMULATIVE SPHERICAL VOLUME AS A FUNCTION OF PORE SIZE

PNL-3 SERIES
PNL-3-27-F SECTION

TYPICAL
AS FABRICATED

*140 μ

390 μ

890 μ

1390 μ

2010 μ

EBR-II IRRADIATED
27,200 MWd/MTM AT 5.0 kW/ft

*DISTANCE FROM FUEL PELLET-CLAD
INTERFACE ARE SHOWN IN MICRONS

HEDL 7605-109.18

WASTAGE ALLOWANCES FOR FTR FUEL PIN DESIGN

TRANSIENT PERFORMANCE ASPECTS

PHENOMENOLOGICAL ISSUES IN FUEL BEHAVIOR UNDER ACCIDENT CONDITIONS

I AXIAL EXPANSION

- EFFECTS OF FUEL CRACKING AND PLASTICITY
- RESTRAINT BY CLADDING

II EARLY FUEL DISPERSAL BY FISSION GASES

- TIME OF INCEPTION AND RATE OF FUEL MOTION
- QUANTITY & DISTRIBUTION OF GASES
- MODE OF FUEL DISRUPTION
- FLOW REGIME FOLLOWING DISRUPTION

III CLADDING FAILURE AND FUEL MOTION

- TIME, LOCATION & EXTENT OF CLADDING FAILURE
- FUEL & SODIUM MOTION AFTER FAILURE
- FISSION GAS AVAILABILITY
- FUEL-SODIUM HEAT TRANSFER

IV STEADY STATE CHARACTERIZATION

TRANSIENT PERFORMANCE MODEL PARAMETERS

FUEL PIN TRANSIENT PERFORMANCE CODES

LIFE - III T

BEHAVE - SST

DEFORM - II

PECTPIN

DILATE/DSTRESS

LAFM

FRUMP - UK

BREDA II - GERMANY

PHYSURA - FRANCE

PIPER - JAPAN

UNCLE - T

LOF CLADDING LOADING AND TEMPERATURES

1900-2400°F TYPICAL CLADDING FAILURE TEMPERATURES

EFFECT OF IRRADIATION ON THE FAILURE STRAIN OF 20% C.W. TYPE 316 STAINLESS STEEL CLADDING DURING HIGH TEMPERATURE TRANSIENTS

REDL 7602-21.1

FUEL PIN CLADDING FAILURE TEMPERATURES DURING HIGH TEMPERATURE LOSS-OF-FLOW TRANSIENTS

HEDL 7602-21.2

LOF SAFETY MARGIN

TOP CLADDING LOADING AND TEMPERATURES

HEATING
RATES

50°F/SEC

LOADING MECHANISM

CLADDING
200°F/SEC

FUEL
1000°F/SEC

- DIFFERENTIAL THERMAL EXPANSION
- INTERGRANULAR FISSION GAS RELEASE
- FISSION GAS INDUCED FUEL SWELLING
- VOLUMETRIC EXPANSION ON FUEL MELTING

FUEL-CLADDING MECHANICAL INTERACTION:
HOOP σ TO 100 ksi

1300-1600°F TYPICAL CLADDING
FAILURE TEMPERATURES

TRANSIENT TESTING VESSEL ASSEMBLY

HEDL MARK-III TREAT SODIUM TEST LOOP

MARK-IIIC LOOP TEST TRAIN

TRANSIENT OVERPOWER TEST SERIES TEST PIN POWER RELATIONSHIPS

SUMMARY OF CAPSULE TESTS (HUT)

<u>EXPERIMENT</u>	<u>FUEL PIN</u>	<u>STEADY STATE</u>		<u>TYPE TRANSIENT (UNTERMINATED)</u>	<u>PIN CLADDING BREACHED</u>
		<u>PEAK POWER KW/FT</u>	<u>PEAK BURNUP MWd/MTM</u>		
HUT CAL-1	PNL 9-69	--	--	CALIBRATION	--
HUT CAL-2	SPECIAL	--	--	CALIBRATION	YES
HUT 5-3A	PNL 10-17	8.8 (AVG)	50,000	50¢/SEC	YES
HUT 5-5A	PNL 9-25	5.6 (LOW)	47,000	50¢/SEC	NO
HUT 5-1A	PNL 11-28	12.4 (HIGH)	47,000	50¢/SEC	NO
HUT 3-3A	PNL 10-20	8.9 (AVG)	51,000	3\$/SEC	NO
HUT 3-5A	PNL 9-45	6.1 (LOW)	51,000	3\$/SEC	NO
HUT 5-5B	PNL 9-34	5.7 (LOW)	48,000	50¢/SEC	YES
HUT 3-1A	PNL 11-47	12.2 (HIGH)	47,000	3\$/SEC	YES
HUT 5-1B	PNL 11-15	12.2 (HIGH)	46,000	50¢/SEC	YES
HUT 3-5B	PNL 9-54	5.8 (AVG)	47,000	3\$/SEC	YES
HUT 3-3B (HOP 3-3C)	PNL 17-34	10.0	30,000	3\$/SEC	YES

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TRANSIENT OVERPOWER TEST SERIES (HUT)

DATA SUMMARY:

<u>TEST</u>	<u>RAMP RATE</u>	<u>T_{CLAD} MAX</u>	<u>MEASURED INCREMENTAL ΔD, %</u>		<u>PIN FAILED ?</u>	<u>TRANSIENT GAS RELEASE, %</u>	<u>MOLTEN FUEL, %</u>
			<u>AVG</u>	<u>MAX</u>			
HUT 5-3A	50¢/ SEC	1514	--	--	YES	--	47*
5-5A	50¢/ SEC	1477	0.1	0.24	NO	14	31
5-1A	50¢/ SEC	1548	0.22	0.55	NO	3	55
3-3A	3¢/ SEC	1448	0.5	0.6	NO	8	50
3-5A	3¢/ SEC	1573	0.5	0.6	NO	5	0
5-5B	50¢/ SEC	1714	--	--	YES	--	49*
3-1A	3¢/ SEC	1525	--	--	YES	--	73*
3-5B	3¢/ SEC	1643	0.4	--	YES	--	61*
5-1B	50¢/ SEC	1682	--	--	YES	--	65*
3-3B(3-3C)	3¢/ SEC	1390	--	0.4	YES	--	34*

*CALCULATED AREAL MELT FRACTION AT TIME OF FUEL PIN FAILURE

Frame 6

Frame 7

Frame 8

Frame 9

PIN HOLE CAMERA PHOTOS

CORRELATION OF FUEL PIN CLADDING FAILURE TIME BY THERMOCOUPLE AND PIN HOLE CAMERA

TESTING APPROACH WITH FUEL CLADDING TRANSIENT TESTER (FCTT)

IRRADIATION-REDUCED FAILURE STRENGTH DEPENDS
ON CLADDING HEATING AND STRAIN RATE

FCTT RESULTS FOR TOP ANALYSIS

CLADDING DUCTILITY REDUCED BY HELIUM EMBRITTLED
INTERGRANULAR FRACTURE ABOVE 1000-1200°F

FCTT FAILURE STRAIN AT $10F^0 / S$

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FCTT FAILURE STRAIN AT 200F°/S

COMPARISON OF TRANSIENT FAILURE STRENGTH FOR FUELED, UNFUELED AND PRESSURIZED IRRADIATED CLADDING

FUEL ADJACENCY EFFECT ON CLADDING MECHANICAL PROPERTIES

FCTT

COMPARISON OF TREAT TESTS WITH FCTT RESULTS

<u>TREAT TEST</u>	<u>FUEL PIN</u>	<u>APPROXIMATE FAILURE CONDITIONS</u>	<u>FCTT CLADDING RESULTS</u>	
			<u>FAILURE STRAIN</u>	<u>FAILURE STRESS</u>
<u>HOP 3-3C</u>	<u>PNL-17-34</u>	1400°F	0.29%	55,000 psi HOOP STRESS
3\$/SEC	30,000 MWd/MTM 700-1100°F 2.4×10^{22} n/cm ²	0.35% STRAIN	(0.2 - 0.5%)	
<u>H5</u>	<u>PNL-17-25</u>	1480°F	0.29%	48,000 psi HOOP STRESS
50\$/SEC	30,000 MWd/MTM 700-1100°F 2.4×10^{22} n/cm ²	0.35% STRAIN	(0.2 - 0.5%)	

ALL STRAIN VALUES ARE THE INCREMENTAL TRANSIENT STRAIN, AND DO NOT INCLUDE THE STRAIN WHICH OCCURRED DURING THE STEADY-STATE EBR-11 IRRADIATION

COMPARISON OF FCTT DATA FOR SPECIMENS FROM TREAT-TESTED PINS WITH SAMPLES FROM STEADY-STATE PINS

CUMULATIVE STRAIN DURING TREAT AND FCTT TESTS

	INCREMENTAL TREAT TEST STRAIN	INCREMENTAL FCTT TEST FAILURE STRAIN	TOTAL TREAT AND FCTT STRAIN	AVERAGE FCTT FAILURE STRAIN*
HOP 3-2A P-23A-27:C	0	0.22%	0.22%	0.30%
HOP 3-2B P-23A-30:D	0.04%	0.13%	0.17%	0.14%

*TESTED AT 10F⁰ / SEC AND 4000 psi GAS PRESSURE

CORRELATION OF STRESS RUPTURE, BURST, TENSILE AND TRANSIENT
STRENGTH DATA WITH THE LARSON-MILLER PARAMETER FOR
20% C.W. TYPE 316 STAINLESS STEEL

THE ROLE OF RETAINED FISSION GASES

INTRAGRANULAR

- INDUCES SWELLING BEFORE FUEL MELTING
- FURTHER EXPANSION WITH FUEL MELTING
- PRODUCES FOAM OR FROTH OF MOLTEN FUEL
- COLLECTS AT GRAIN BOUNDARIES

INTERGRANULAR

- GRAIN BOUNDARY SEPARATION
- CONTRIBUTES TO FROTHING OF MOLTEN FUEL
- CONTRIBUTES TO INTERNAL PRESSURE
- MAY BE LOST TO SYSTEM

CENTRAL VOID GAS

- INTERNAL PRESSURIZATION
- DRIVING FORCE FOR MOLTEN FUEL
- MAY BE LOST TO SYSTEM

FISSION GAS BEHAVIOR STUDIES

- QUANTITATIVE METALLOGRAPHY
- DIRECT ELECTRICAL HEATING
- EXTERNAL ELECTRICAL HEATING

RESULTS

- RETAINED GAS DISTRIBUTION
- DIRECT OBSERVATION OF FUEL BEHAVIOR
- RELEASE BEGINS AT $\sim 2000^{\circ}\text{C}$
- FRAGMENTATION OCCURS BELOW MELTING
- FILLS RADIAL
- RADIAL FUEL EXPANSION
- RAPID HEATING CAUSES DISPERSAL
- SLOW HEATING CAUSES SWELLING

FISSION GAS BEHAVIOR IN MIXED OXIDE FUEL "UNRESTRUCTURED" FUEL ZONE

AFTER STEADY STATE IRRADIATION

AFTER TRANSIENT OVERPOWER IRRADIATION

INTRAGRANULAR POROSITY DEVELOPMENT

INTERGRANULAR POROSITY DEVELOPMENT

RADIAL DISTRIBUTION OF "INTRAGRANULAR" XENON

HEDL 7703-152

FISSION GAS BEHAVIOR MODELS

FRAS CODE

- CALCULATES RATE OF RELEASE FROM GRAINS TO GRAIN BOUNDARIES
- CALCULATES SWELLING OF GRAINS
- RANDOM & BIASED MIGRATION OF BUBBLES TREATED

PFRAS

- CORRELATION FOR ACCIDENT ANALYSES
- $F = a_1 \nabla T \exp \left\{ -a_2/T - (a_3 + a_4 T) F \right\}$

EMPIRICAL CORRELATION
OF FUEL FAILURE THRESHOLDS

- CORRELATES CONDITIONS AT TIME AND LOCATION OF FUEL PIN CLADDING FAILURE FOR PROTOTYPIC TREAT TESTS.
- USED TO PREDICT TIME AND LOCATION OF FUEL PIN CLADDING FAILURE IN MELT ACCIDENT ANALYSIS CODE.

HEDL 7604-232

FIGURE 1. Failure Potential vs Local SS Pin Power

REVISED EMPIRICAL CORRELATION

$$FP = a \left(\frac{\Delta H}{Y} \right) \text{time avg } t^b \exp(cs) D^d$$

FP = FAILURE POTENTIAL

ΔH = ENTHALPY UPSET OF NON-COLUMNAR GRAIN REGION, J/gm

Y = MATERIAL STRENGTH AS DEFINED BY FCTT TESTS, psi

t = TIME INTO TRANSIENT

S = EXPRESSION OF FUEL STRUCTURE, FUNCTION OF VOLUME FRACTION OF NON-COLUMNAR GRAIN AND FUEL-CLADDING GAP AT STEADY STATE

D = TOTAL AVAILABLE VOID VOLUME AS FABRICATED

a, b, c, d = ADJUSTABLE CONSTANTS

ALTERNATE FUELS - KEY SAFETY ISSUES

FUEL BEHAVIOR

- OXIDES - SIMILAR TO (U, Pu) O₂
- CARBIDES/NITRIDES - FISSION GAS RETENTION/RELEASE
 - FUEL SWELLING
 - FCMI
 - MFCI
- METALS - EUTECTIC FORMATION WITH CLADDING

PIN FAILURE PROPAGATION

- OXIDES - SIMILAR TO (U, Pu) O₂
- CARBIDES/NITRIDES - PROBABLY UNLIKELY
- METALS - NOT LIKELY

ALTERNATE FUELS - KEY SAFETY ISSUES

SUBASSEMBLY FAULT PROPAGATION

- OXIDES - SIMILAR TO (U, Pu) O₂
- CARBIDES/NITRIDES - GREATER THAN (U, Pu) O₂
 - DELAYED FUEL DISPERSAL
 - GREATER PROPENSITY FOR FREEZING & PLUGGING
 - GREATER POTENTIAL FOR MFCI
- METALS - EUTECTIC FORMATION INCREASES POTENTIAL

RESPONSE TO WHOLE CORE ACCIDENT INITIATORS

LOF

- OXIDES - SIMILAR TO (U, Pu) O₂
- CARBIDES/NITRIDES - POTENTIAL FOR FUEL COLLAPSE OFFSET BY FISSION GAS RETENTION
- METALS - SIMILAR TO CARBIDES/NITRIDES

TOP

- OXIDES - SIMILAR TO (U, Pu) O₂
- CARBIDES/NITRIDES - HIGHER CONDUCTIVITY: INCREASED POTENTIAL PLUGGING, FREEZING, EXTENDED COOLING
- METALS - SIMILAR TO CARBIDES/NITRIDES

SAFETY CONSIDERATIONS IN FAST REACTOR FUEL DEVELOPMENT - SUPPLEMENT

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SAFETY CONSIDERATIONS IN FAST REACTOR FUEL DEVELOPMENT* - SUPPLEMENT
IAEA-CN-36/18 (IV.2)

In the paper prepared for this conference we presented an overview of how fast reactor safety has been addressed in the U.S. fast reactor fuel development program and how the results of these studies influenced fast reactor safety analyses. Two principal issues in fuel behavior appear: early fuel dispersal by fission gas in voided subassemblies during a loss-of-flow (LOF) accident and fuel pin cladding failure and subsequent fuel motion in subassemblies subjected to severe power transients while the cladding is still relatively cold and strong. The latter concern arises during the full-flow transient overpower (TOP) accident and in partially-voided or unvoided subassemblies during a voiding-driven power excursion in the LOF case.

This presentation will focus on some important aspects of fuel pin behavior phenomenology up to the point of cladding failure during hypothetical core disruptive accidents (HCDA).

From our analyses and transient tests of stainless steel clad mixed oxide fuel pins in TREAT we have identified four fuel pin cladding failure causes:

SLIDE 1

Differential fuel-cladding thermal expansion refers to the mechanical stressing of the cladding due to greater thermal expansion of the fuel.

Fission gas release refers to direct hydrostatic stressing of the cladding caused by the release of fission gases during events which cause a temperature excursion in the fuel.

Volumetric expansion of the fuel refers to mechanical stressing of the cladding due to volume increase of the fuel caused by precipitation and growth of fission gas bubbles in the fuel matrix and of course, the approximately ten percent increase in fuel volume upon melting.

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Fission gas pressure refers to hydrostatic stressing of the cladding due to the ever-increasing plenum pressure from accumulation of gases during steady-state operation.

To illustrate how these failure causes interact, let us consider the two classic HCDA's; loss of coolant flow with failure to scram and transient overpower with failure to scram.

SLIDE 2

As illustrated in this slide, the flow coastdown phase of the unprotected loss of coolant flow accident is typified by rapidly increasing cladding temperatures and eventually coolant boiling. Cladding failure temperatures in the voided subassemblies are generally in the 1038°C to 1316°C range; the higher failure temperature representing that for fresh fuel. In these subassemblies the cladding expands away from the fuel so there is no fuel-cladding mechanical interaction and cladding failure is due to the increasing plenum pressure and decreased strength of the cladding at elevated temperatures.

SLIDE 3

In LMFBR fuel pins, cladding ballooning and subchannel blockage will not occur during the flow coastdown phase of the LOF accident. As this slide illustrates, in the unirradiated condition the 20 percent cold worked cladding can survive large strain. However, the cladding deformation cause, fission gas pressure, is not present so cladding failure occurs near the melting point of the steel with little strain. In the irradiated condition, fission gas release has increased the internal pressure and the cladding has been embrittled by radiation damage processes. Under these conditions failure strains are much less than those required for subchannel blockage.

Post-failure behavior of the fuel is characterized in current LOF analyses by fuel dispersal due to the combined effects of expansion of the fission gases in the fuel and eventually, fuel vapor pressure. Even if not dispersive, fission gas release is a significant factor in mitigating accident energetics by preventing fuel compaction.

From the foregoing it is expected that behavior of the fuel pin in the initial phases of a LOF HCDA would be relatively insensitive to the usual fuel pin design parameters, i.e., pin diameter, cladding thickness and material, fuel smeared density, etc.

Fuel pin behavior in transient overpower conditions is significantly more complicated.

SLIDE 4

This slide illustrates that for the full-flow, unprotected transient overpower accident the rate of temperature increase in the fuel is substantially greater than in the flow coastdown phase of the LOF-HCDA accident and the temperature of the cladding at the time of failure is generally less; 704°C to 871°C. In the LOF-driven TOP the cladding failure temperatures may be substantially higher. Differential fuel-cladding thermal expansion mechanically stresses the cladding. The rapidly increasing fuel temperatures cause fission-gas-stressing of the cladding. When fuel melting occurs, the phase change volume increase also stresses the cladding. Our tests and analyses show that differential thermal expansion is the predominant cause of cladding stress but its effect is mitigated by plasticity, creep and hot pressing of the fuel. Understandably, this stressing mechanism is extremely sensitive to the residual fuel-cladding gap and manifests itself in the following manner: if the hot (operating) gap is very small, differential expansion stressing of the cladding will occur early in the transient. This would cause breach of the cladding with little molten fuel present. Subsequent fuel ejection and sweepout of the fuel would result in shutdown or hydraulic disassembly of the reactor. If the hot gap is large, stressing of the cladding will occur later in the transient, providing more time for greater energy generation and consequently larger volumes of molten fuel at the time of cladding failure. Larger volumes of molten fuel increase the propensity for fuel assembly blockage with solidified fuel and steel during sweepout and hydraulic dispersal.

SLIDE 5

The behavior of the fission gas dissolved in the fuel matrix, contained as bubbles in the grains and on the grain boundaries during fuel temperature excursions is important from two standpoints. First, it represents an additional cladding stressing mechanism and secondly, it significantly contributes to the fuel dispersal-sweepout mechanism.

We generally assume that gases released from the fuel during an over-power transient do not have time to equilibrate with the gas plenum, so they immediately exert a local hydrostatic force on the cladding.

Our quantitative ceramography of fuel after a temperature transient shows precipitation and growth of bubbles from gases coming out of solution in the grains and the formation and growth of bubbles at the grain boundaries. Both phenomena cause solid state fuel expansion and cladding stress, as well as contribute to fuel dispersal.

SLIDE 6

This ongoing work, in conjunction with electron microprobe analyses of the relative radial distribution of intragranular xenon will better quantify the fission gas and fuel expansion components of cladding stress during the TOP-HCDA accident. An interesting feature illustrated in this slide is that the xenon distribution in the fuel is not as uniform as we once believed. This sample was taken from a low linear power fuel pin after long steady-state irradiation at approximately 18 kW/m (5.5 kW/ft). The rapid decrease in xenon concentration as a function of distance from the outer surface of the fuel explains why the transient overpower fission gas effects in low power fuel are not as pronounced as we had earlier anticipated. In fuel with higher linear heat ratings the disparity between our earlier assumption for radial gas distribution and that typified here decreases with increasing linear power since the unrestructured region of the fuel pellet is smaller in the higher linear power fuel. Gas distribution, as illustrated here, combined with a larger hot fuel-cladding gap in low linear power fuel pins tends to decrease the time interval between cladding failure of low linear power and high linear power fuel during a transient overpower HCDA.

(SLIDE OFF)

Our studies of the transient performance of stainless steel clad mixed oxide fuel pins also include determination of the mechanical properties of the cladding under simulated transient overpower or loss-

of-flow conditions. This work has provided significant understanding of fuel pin cladding mechanical properties under dynamic loads as a function of fluence, temperature and, interestingly, adjacency to fuel. This point is illustrated in the next slide.

SLIDE 7

The ordinate is the ratio of the stress at failure of irradiated cladding to that of unirradiated cladding. The small difference between unfueled, stressed, and unfueled, unstressed, irradiated cladding is not considered statistically significant. Note that these data show irradiation creep to be nondamaging. The significant information in this graph is that cladding samples taken from fuel pins show a substantial decrease in stress ratio over the fluence range tested thus far. We call this phenomenon the fuel-adjacency effect and we currently attribute it to helium embrittlement of the inner regions of the cladding wall as a result of ternary fission events. Prediction of time and location of fuel pin cladding failure must carefully consider the effect of this phenomenon. This work also illustrates that mechanical properties data from structural materials irradiations may not be directly applicable to fuel pin design and performance analyses, particularly if the design bases include consideration of transient performance limits.

Time and location of fuel pin cladding failure in a transient overpower HCDA are more sensitive to fuel pin design parameters than in the case of the LOF HCDA. For example, higher fuel smeared densities should promote earlier failure of the cladding. Higher density pellets would retain more fission gases. Smaller fuel-cladding diametral gaps would promote earlier cladding failure, and so on.

In summary, the combined testing and analysis program for the development of Liquid Metal Fast Breeder Reactor mixed oxide fuel continues to augment the data base from which well-founded mechanistic analyses of fuel design, performance and safety can be performed.

FUEL PIN CLADDING FAILURE MECHANISMS UNDER ACCIDENT CONDITIONS

- **DIFFERENTIAL FUEL-CLADDING THERMAL EXPANSION**
- **RAPID FISSION GAS RELEASE**
- **VOLUMETRIC EXPANSION OF FUEL**
 - **FISSION GAS BUBBLE EXPANSION**
 - **MELTING**
- **FISSION GAS PRESSURE**

LOF CLADDING LOADING AND TEMPERATURES

1040-1315°C TYPICAL CLADDING FAILURE TEMPERATURES

EFFECT OF IRRADIATION ON THE FAILURE STRAIN OF 20% C.W. TYPE 316 STAINLESS STEEL CLADDING DURING HIGH TEMPERATURE TRANSIENTS

TOP CLADDING LOADING AND TEMPERATURES

HEATING
RATES

50¢/SEC

CLADDING
111C°/SEC

FUEL
555C°/SEC

LOADING MECHANISM

- DIFFERENTIAL THERMAL EXPANSION
- INTERGRANULAR FISSION GAS RELEASE
- FISSION GAS INDUCED FUEL SWELLING
- VOLUMETRIC EXPANSION ON FUEL MELTING

FUEL-CLADDING MECHANICAL INTERACTION:
HOOP σ TO 100 ksi

700-870°C TYPICAL CLADDING
FAILURE TEMPERATURES

FISSION GAS BEHAVIOR IN MIXED OXIDE FUEL "UNRESTRUCTURED" FUEL ZONE

AFTER STEADY STATE IRRADIATION

AFTER TRANSIENT OVERPOWER IRRADIATION

INTRAGRANULAR POROSITY DEVELOPMENT

INTERGRANULAR POROSITY DEVELOPMENT

HEDL-7704-6

RADIAL DISTRIBUTION OF "INTRAGRANULAR" XENON

COMPARISON OF TRANSIENT FAILURE STRENGTH FOR FUELED, UNFUELED AND PRESSURIZED IRRADIATED CLADDING

IAEA-CN-36/18 (IV.2)

SAFETY CONSIDERATIONS IN FAST REACTOR FUEL DEVELOPMENT

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SAFETY CONSIDERATIONS IN FAST REACTOR FUEL DEVELOPMENT

The purpose of this paper is to discuss how fast reactor safety has been addressed in the U.S. fast reactor fuel development program and how the results of these studies have influenced fast reactor safety analyses.*

Fast reactor safety philosophy in the U.S. as expounded by reactor developers, designers and the licensing and regulatory agencies is the subject of several recent papers [1, 2, 3, 4]. Fuel development is conducted in the context that there are four lines of assurance (LOA) [1] to protect the public and the environment:

LOA 1	Prevent Accidents
LOA 2	Limit Core Damage
LOA 3	Contain Accidents in Inner Plant Barrier
LOA 4	Attenuate Radiological Products Dispersal

The objectives of work in support of LOA-1 and -2 are to demonstrate the low probability of a core disruptive accident (CDA) and the objectives of work in support of LOA-3 and -4 are to demonstrate that the consequences of a CDA are adequately limited by the plant containment capability. LOA's -1, -2 and -3 have the greater influence on the fast reactor fuel development program. LOA-4 is not discussed in this paper since this safety issue has no strong influence on fuel pin design and development.

LOA-1 emphasizes reliability and the intrinsic ability of a well designed fuel pin to prevent propagation of a local fault into an accident initiator. Fuel pin design objectives (Table I) include maintaining cladding integrity into the hypothetical event range. Pin performance during protected emergency and faulted events as well as hypothetical events terminated by the plant-protection-system (PPS) are considered in detail [5], as are the more usual fuel pin design considerations for steady state fuel burnup capability without undue cladding deformation such as caused by fuel-cladding mechanical interaction, fission gas pressure and thermal and neutron-irradiation-induced-creep and swelling of the cladding.

Steady state irradiation testing to demonstrate reliability and design adequacy of the Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor (CRBR) mixed oxide fuel pins is well documented [6, 7]. Performance limits established in the design of the FFTF and CRBR fuel pin influenced the PPS design with respect to speed of response and control rod worth. For example, one FFTF fuel pin design requirement was that incremental cladding strain due to normal and upset events (i.e., normal start-ups, normal and emergency shutdowns) should not exceed 0.1 percent. Analytic iteration between reactor system response to rapid emergency shutdown and fuel pin cladding deformation was used to define PPS system requirements. These analyses also defined boundary events for proof testing the fuel pin and the PPS design bases (Table I). In the Engineering Proof Tests, prototypic fuel pins previously irradiated in the Experimental Breeder Reactor II (EBR-II) are subjected in the Transient Reactor Test Facility (TREAT) to the thermal equivalent of the FFTF or CRBR upset event. This is followed

* This work performed under the auspices of the U.S. ERDA Division of Reactor Development and Demonstration

by detailed postirradiation examination to confirm the absence of damage to the pin. In the transient irradiation tests which directly support CRBR project, the pins are subjected in TREAT to multiple simulated upset events. Several tests require subsequent steady state irradiation in EBR-II until cladding failure occurs to demonstrate that multiple upset events do not reduce the fuel pin lifetime to less than the design requirement. The first six pins in these tests have been reinserted into EBR-II. Table II summarizes present results from the Engineering Proof Tests. A similar program is being conducted with carbide and nitride fast reactor fuel pins.

A parallel program of Undermined Tests in TREAT subjects previously irradiated (EBR-II) prototypic fuel pins to thermal simulations of FFTF or CRBR emergency and faulted events until incipient cladding failure occurs. These studies experimentally address LOA-2 by quantifying fuel pin behavior characteristics such as; fission gas behavior, cladding mechanical properties under dynamic conditions, time and location of fuel pin cladding failure, fuel motion within the pin and fuel motion out of the cladding breach.

Upper boundary conditions (Table I) which represent the most severe test of fuel pin cladding integrity for the FFTF and CRBR provide bases for further irradiation tests in TREAT. Undercooling events, i.e., the FFTF Continuous Flow Reduction (CFR) and the CRBR Transition to Natural Circulation (TNC) events are classified as emergency events, hence the fuel pin design requirement is to maintain cladding integrity. Simulation of the FFTF CFR event in TREAT is relatively straightforward. However, the time required for simulating the CRBR flow coastdown and TNC event exceeds the single cycle energy capability of TREAT. We believe we can demonstrate cladding integrity and quantify the cladding failure threshold for this event by application of data from (a) out-of-reactor cladding mechanical properties tests [8], (b) Thermal Hydraulic Out of Reactor Safety Facility [1] and (c) The Sodium Loop Safety Facility [1]. Table III summarizes the status of the "Undermined Tests" in the FFTF/CRBR fuel development program. A similar program is being conducted in the parallel carbide and nitride fuel development program [9].

In recognition of the importance of fission gas behavior in fast reactor accident analyses [10, 11] a substantial program is in progress to quantify fission gas behavior during overpower and loss-of-coolant events. Quantitative metallography techniques are being used to determine axial and radial distribution of fission gases in the fuel before and after transient testing. These data provide a broader base for refined calibration and verification of the fission gas behavior FRAS code [12]. Codes for the analysis of fuel pin mechanical behavior under transient conditions (such as BEHAVE [13], PECTPIN and DILATE/DSTRESS [14]) will use the improved FRAS model for treatment of fission gas behavior.

Accident analyses for LOA-2 and LOA-3 for LMFBFR's are based on two major initiators: the unprotected Transient Overpower (TOP) and Transient Undercooling (TUC) events. Development of two major code systems for analyzing these events resulted in MELT-IIIA [15] and SAS-3A [16]. Fuel pin failure time and location is currently predicted by the Failure Potential Correlation [17] which has been developed from the results of twelve TREAT tests. A parallel effort is underway to develop mechanistic models for fuel pin transient performance [13, 14, 18].

MELT was first applied in the LMFBFR TOP analyses using parametric fuel motion models [19, 20]. We are now modifying the MELT code by incorporating more mechanistic fuel motion models. Recently the Hydrodynamics

of Two Phase Internal Motion (HOTPIM) model [21] was developed and incorporated into the MELT system. HOTPIM provides a more detailed model of internal fuel motion and ejection than does the earlier uniform cavity model. More recent mechanistic models for fuel-coolant interaction and for motion of fuel in the coolant channel under TOP conditions have been developed and incorporated in the MELT code [10]. These phenomena have also been modeled by the PLUTO-1 code and SAS/FCI module, and more recently by the PLUTO-2 code [11]. Current effort is to develop a non-uniform interaction region model for incorporation into the MELT code. This effort will culminate in a mechanistic fuel-cladding plateout/ablation and blockage model. The combined effect of these developments demonstrates a stronger negative reactivity feedback caused by the movement of fuel out of the active core region, leading to early termination of the accident. Investigations of the noncoherent failure of fuel pins strongly suggest that plugging of the 217 pin subassembly is less probable than in the coherent simultaneous failure sequence presently assumed in the MELT multi-channel calculations [11].

Application of the SAS code to analyses of TUC events has led to identification of early fuel dispersal as the key issue in the energetics of such postulated accidents in CRBR. FRAS and other new models show a greater potential for fuel dispersal in the whole reactor case than indicated from the single-pin, dry-capsule TREAT tests F1 and F2 [18]. SAS is also being extended by the FUMO model [27] to represent two-phase molten fuel motion in fuel assemblies to assess the potential dispersive mechanism. Incorporation of additional models into the SAS code system resulted in the SAS-4A code [11]. This will increase the flexibility of SAS for investigating TUC events leading to core disruption for a variety of LMFBR designs. Inclusion of PLUTO-2 is a major improvement which provides better capability for mechanistic analysis of fuel ejection and subsequent fuel motion in an unvoided or a partially voided coolant channel. Such conditions are expected in the Transient Undercooling-driven-Overpower (TUCOP) events postulated for large LMFBR cores with positive void reactivity coefficients [18].

Events leading to a totally disrupted core are classified as extremely unlikely or hypothetical by the majority of LMFBR designers. Nevertheless, consequences of such postulated events are analyzed as an additional safeguard against release of hazardous materials to the biosphere. To this end, a major effort has been mounted toward developing techniques to describe the transition and disassembly phase of the TUC accident should this description be necessary. The PAD [28] and Venus [22] codes have been developed for neutronic-hydrodynamic analyses of hypothetical, short-period, possibly energetic power excursions. Comparable calculations [25] (PAD in spherical and Venus in finite-cylindrical coordinates) show good agreement and testing of these codes against experiments is satisfactory. Significantly, these codes predict substantially less damage energy than do previously used analytical techniques. Recent PAD Code analysis [25] of the classical Bethe-Tait hazard study [26] of the Dounreay reactor shows that the kinetic energy transmitted to the vessel head should be reduced by a factor of at least 25. Nearing operational status, the SIMMER [23, 24] code is an advanced 2-dimensional, multi-material code for disrupted core analysis. It will include more parametric details than possible in existing codes.

In summary, we have described the strong interrelationship among fast reactor system designers, safety analysts and fuels research and development activities in the U.S. LMFBR Program. Through the synergism of this integrated effort substantial progress has been made in understanding and mechanistically modeling the phenomena which govern the mechanical behavior of fast reactor cores under postulated accident conditions. It is extremely encouraging that as experimental data have been obtained and models further refined, the calculated probabilities and consequences of fast reactor core disruptive accidents become progressively smaller and increasingly tractable from the designer's standpoint.

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TABLE I
LMFBR FUEL PIN DESIGN REQUIREMENTS

Category	Type Event	Requirements	Examples
Normal Operation	Steady State	Maintain Cladding Integrity to Design Lifetime	Steady State Operation
	Operational Transients		Startup and Shutdown
Upset	Operational Incident	Maintain Cladding Integrity to Design Lifetime	CRBR-U16 (30¢ Step, Overpower) FFTF-U10 (3¢/sec. Overpower)
Emergency	Minor Incident	Maintain Cladding Integrity For One Event	CRBR-U18S (Loss of Power, Undercooling) -E16 (3-Loop Natural Circulation, Undercooling) FFTF-CFR (Continuous Flow Reduction, Undercooling)
Faulted	Major Incident	Maintain Coolable Geometry For One Event	CRBR-F1 (60¢ Step, Overpower) CRBR-2\$/sec. (PPS Design Transient, Overpower) FFTF-E6S (3\$/sec., Overpower, PPS Design Transient)
	Hypothetical Events	Evaluate Consequences	FFTF HCDA CRBRP HCDA

TABLE II. ENGINEERING PROOF TEST RESULTS

EXPERIMENT	STEADY STATE				TRANSIENT		
	LINEAR HEAT RATING kW/M	CLADDING TEMPERATURE @MIDPLANE, °C	FUEL BURNUP, MWd/kg	TOTAL ENERGY DEPOSITED J/g	PEAK INSTANTANEOUS POWER kW/M	MAXIMUM CLADDING TEMPERATURE, °C	MAXIMUM INCREMENTAL CLADDING STRAIN %
HOP 3-3B*	34.1	453	30.9	1395	152.2	566	0
HOP 3-1A	35.1	506	7.3	1414	169.5	593	0
HOP 3-2A	38.4	583	21.4	2351	292.3	738	0.4
HOP 3-2B	35.1	565	42.1	2030	241.5	704	0.4

*Signifies Simulated 3\$/SEC Transient

TABLE III - UNTERMINATED TEST RESULTS

EXPERIMENT	STEADY STATE				TRANSIENT					
	Linear Heat Rating kW/M	Cladding Temp. @ Midplane, °C	Fuel Burn- up, MWd/kg	Total Energy Deposited, J/g	Peak Inst. Power, kW/M	Maximum Cladding Temp. °C	Maximum Incremental Cladding Strain %	Cladding Failed?	Transient Gas Release, %	Areal Fuel Melting %
HUT 5*-3A	28.9	482	50	2765	187.8	824	-	Yes	30	47†
HUT 5-5A	18.2	468	47	2433	137.3	802	0.24	No	15	31
HUT 5-1A	40.6	506	47	3071	187.9	843	0.55	No	4	55
HUT 3-3A	29.3	484	51	1716	522.5	788	0.9	No	9	50
HUT 3 ⁺ -5A	19.9	477	51	1363	435.2	857	0.6	No	5	0
HUT 5-5B	18.7	474	48	2926	179.5	935	-	Yes	8	49†
HUT 3-1A	40.0	511	47	2070	538.0	829	-	Yes	25	73†
HUT 3-5B	19.0	463	47	1547	430.1	891	-	Yes	15	61†
HUT 5-1B	40.0	511	46	3488	221.2	916	-	Yes	30	65†

• Signifies simulated 50¢/s transient

+ Signifies simulated 3\$/s transient

† Calculated at time of cladding failure