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PERFORMANCE OF REFRACTORY ALLOY-CLAD FUEL PINS

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C. M. Cox
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PERFORMANCE OF REFRACTORY ALLOY CLAD FUEL PINS

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This paper (Figure 1) will discuss objectives and basic design of two fuel-cladding tests being irradiated in support of SP-100 technology development. Two of the current space nuclear power concepts use conventional pin type designs, where a coolant removes the heat from the core and transports it to an out-of-core energy conversion system. An extensive irradiation testing program was conducted in the 1950's and 1960's to develop fuel pins for space nuclear reactors. The program emphasized refractory metal clad uranium nitride (UN), uranium carbide (UC), uranium oxide (UO_2), and metal matrix fuels (UCZr and BeO-UO_2). Based on this earlier work, our studies concluded that UN and UO_2 fuels in conjunction with several refractory metal cladding materials showed high potential for meeting space reactor requirements[2,3,4] and that UC could serve as an alternative but higher risk fuel.

Weaver and Scott,[5] Gluyas and Watson,[6] and Mayer[7] et al., provided summaries and experiment descriptions of the irradiation experience with UN, UC, and metal matrix fuels. The UO_2 fuel irradiation data are more dispersed; but Chubb, W. Storhok and Keller[8] and Kangilaski[9] are good starting points. These and other documentation of these early tests provide fuel pin fabrication parameters and performance measurements for these tests. As a result of this effort, a data base of 342 fuel pins having fuel and cladding combinations applicable to SP-100 designs was collected. This data base is summarized in Figure 2. In addition, data were collected on 362 fuel pins having claddings of interest but fuel not used in the current concepts (UCZr and BeO-UO_2).

Particular attention was payed to the relevant irradiation performance data as guided by the SP-100 reactor core requirements of peak fuel burnup up to 5 at.%, fast neutron fluence to 3 x 10^{22} n/cm^2, and coolant temperatures up to 1500 K. The fuel pin irradiation data base relevant to these requirements is summarized in Figure 3 where each point represents a fuel pin.

Fig. 1

PERFORMANCE OF REFRACTORY ALLOY-CLAD FUEL PINS

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SPACE REACTOR FUELS DATA BASE

	UC		UN		UO_2
W-26Re	14	(8)*	33	(14)	27
TUNGSTEN			2		1
T-111					
WITH DIFFUSION BARRIER			55	(14)	
WITHOUT DIFFUSION BARRIER			2	(1)	2
TZM					
WITHOUT DIFFUSION BARRIER			6		
Cb-1Zr					
WITH DIFFUSION BARRIER	47		35		2
WITHOUT DIFFUSION BARRIER	13	(2)	15		35
PWC-11					
WITH DIFFUSION BARRIER			47	(3)	
AUSTENITIC STAINLESS STEELS			6		
SUBTOTALS	74	(10)	201	(32)	67
TOTALS			342		

*NUMBERS IN PARENTHESES INDICATE FAILED PINS

HEDL 8405-074.6

Fig. 2

SPACE REACTOR FUELS DATA BASE

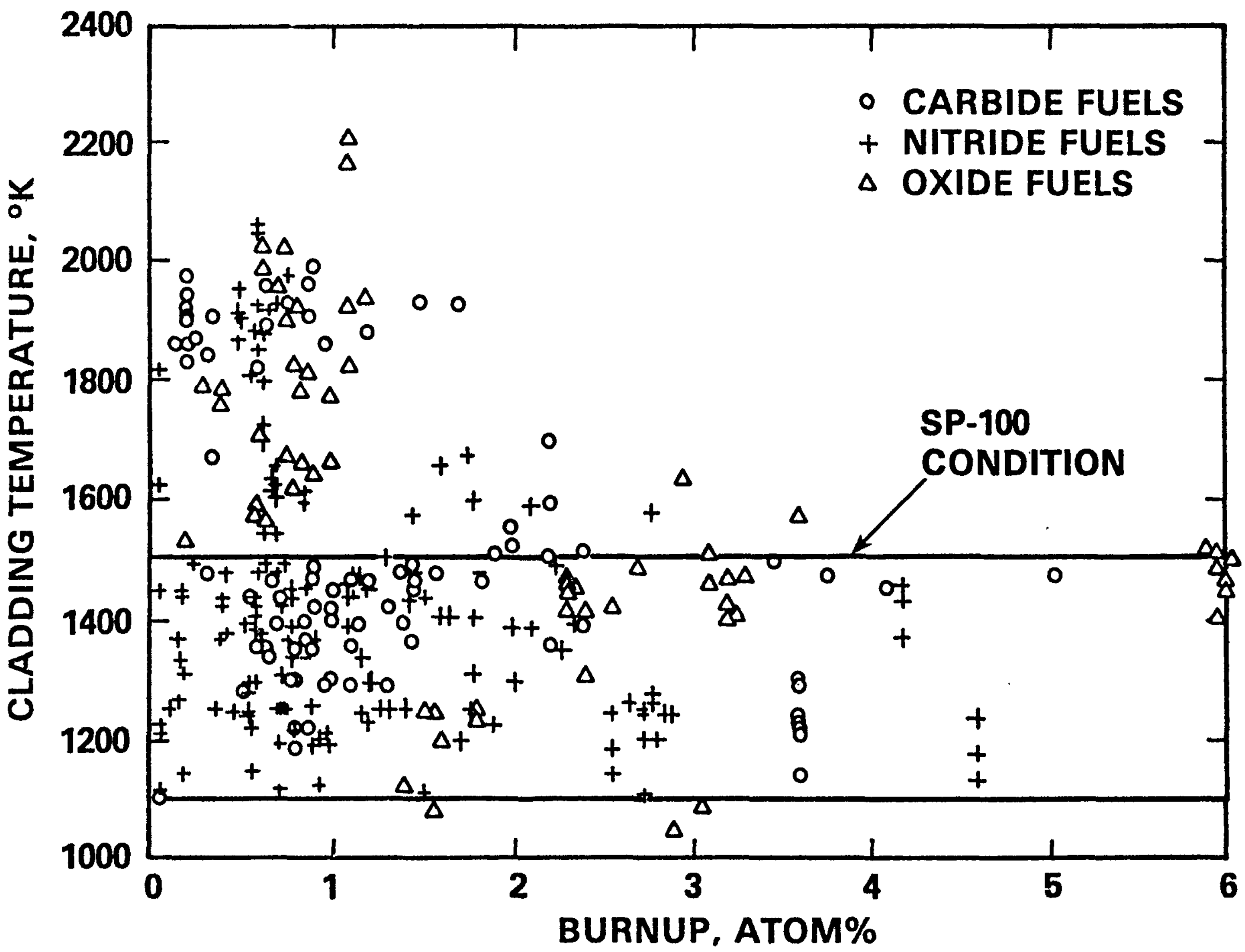

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Fig. 3

Although significant conclusions are drawn from these data, they suffer the nonprototypic feature of having been irradiated in thermal rather than fast neutron reactors; thus, the effects of fast neutrons on cladding properties were not simulated, and the fission rate distribution across the fuel diameter had a marked depression in the center of the fuel. Moreover, the length of the fuel column was quite small in many of the experiments, as shown in Figure 4. This results in highly uncertain predicted temperatures and powers due to complicated heat transfer paths and magnified effects of power peaking at the ends of fuel columns in thermal reactors.

To summarize the experience with refractory metal clad fuel elements: with certain qualifications which can be accommodated by appropriate design, there are a number of fuel/cladding combinations which could meet the current SP-100 performance requirements (Figure 5). Tungsten alloys used with either UN or UO_2 have the best potential to meet both the lifetime and operating temperature requirements for current designs. However, from a systems point of view, there is a weight penalty in using W. Tantalum alloys will work with either UN or UO_2 provided a W diffusion barrier is used with the UN fuel. Because of weight and neutronics considerations Ta alloys are most practical with a system design using UN. Irradiation data with Mo alloys do not exist, but based on compatibility and strength considerations Mo alloys are expected to meet SP-100 performance requirements with either UO_2 or UN fuel. Nb alloys have the lowest creep strength, and thus may require vented fuel pins. They require a W diffusion barrier if used with UN. However, there is more fabrication experience with Nb than any of the other alloys.

The previous irradiation testing programs provided considerable guidance to the selection of fuel and cladding candidates. However each of the fuel/cladding combinations has one or more feasibility issues associated with it that can not be conclusively resolved with the available data. The major issues (Figure 6) are: high temperature UN swelling, fuel/cladding compatibility, fast neutron damage to cladding materials, and UO_2/Li reaction in breached fuel pins.

IRRADIATION DATA BASE

RANGE OF LENGTHS

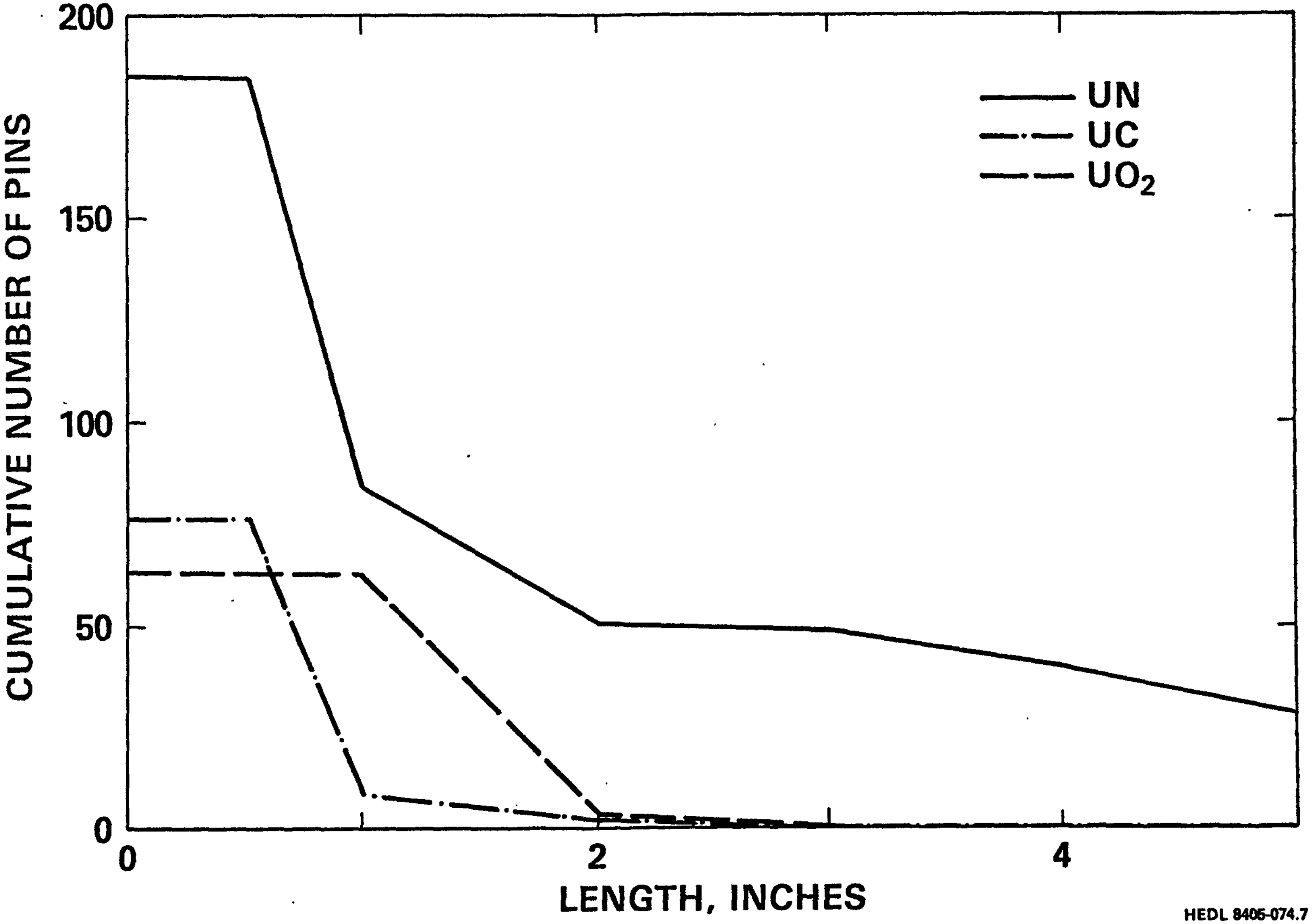


Fig. 4

FUEL AND CLADDING CANDIDATES FOR SP-100 APPLICATION

FUELS	CLADDINGS
UN	W-26Re
	Ta-8W-1Re-0.7Hf-0.025C (ASTAR-811C)
	Mo-13Re
UO_2	Nb-1Zr
	Nb-1Zr-0.1C (PWC-11)

Fig. 5

OUTSTANDING ISSUES

- HIGH BURNUP SWELLING OF UN
- FUEL/CLADDING COMPATIBILITY
- DEVELOPMENT OF DATA BASE WITH PROTOTYPICALLY CONFIGURED FUEL PINS AND FAST NEUTRON IRRADIATION
- EFFECTS OF UO_2-Li REACTION IN EVENT OF PIN FAILURE

Fig. 6

These issues must be addressed by a contemporary testing program to assure the feasibility of such fuel pins under the high temperature, fast neutron flux, and long lifetime conditions of SP-100.

SP-1 and SP-2 have been designed and partially constructed for irradiation in the Experimental Breeder Reactor-II (EBR-II) to address the first three feasibility issues. SP-1 will provide interim examination data to assist in the fuel system selection in July 1985, and both experiments will be completed, to SP-100 goal burnup, by 1987. Irradiation of SP-1 began in December 1984 with fabrication of SP-2 proceeding towards a mid-1985 beginning of irradiation.

The irradiation test vehicle selected for the experiment is a modified version of a proven irradiation test vehicle used for high temperature B_4C irradiations for the LMFBR program. The test vehicle contains four test subassemblies, Figure 7. The outer surfaces of these test subassemblies, Figure 8, are exposed to the reactor sodium coolant at $\sim$650 K. A stainless steel subcapsule is used to increase the temperature to the desired fuel pin cladding temperature, 1300 to 1500 K, by providing two insulating gas gaps Figure 9. The innermost capsule, fabricated from TZM, a molybdenum alloy, is used to contain each fuel pin submerged in lithium. The lower TZM capsule in each assembly is designed so a thermal expansion device, TED, can be included to assist, via postirradiation examinations, in determining operating temperatures. Preirradiation design uncertainties in operating temperature, including the effects of natural circulation in the lithium filled annulus are $\sim\pm$60 K. Postirradiation measurements of the TED's coupled with burnup analyses are expected to reduce this to $\sim\pm$30 K.

Fuel pin fabrication parameters for the experiments, Figure 10, was influenced by the desire to obtain nearly prototypic, but accelerated (relative to the SP-100 seven year lifetime) data, and the constraints of EBR-II. The capsules were designed to accommodate two fuel pins in each of the four test positions; making a total of eight fuel pins in each test vehicle. To fit within this envelope the test fuel pins are 6.4 in. long and contain a 3 in.

SP-100 FUEL TEST ASSEMBLY

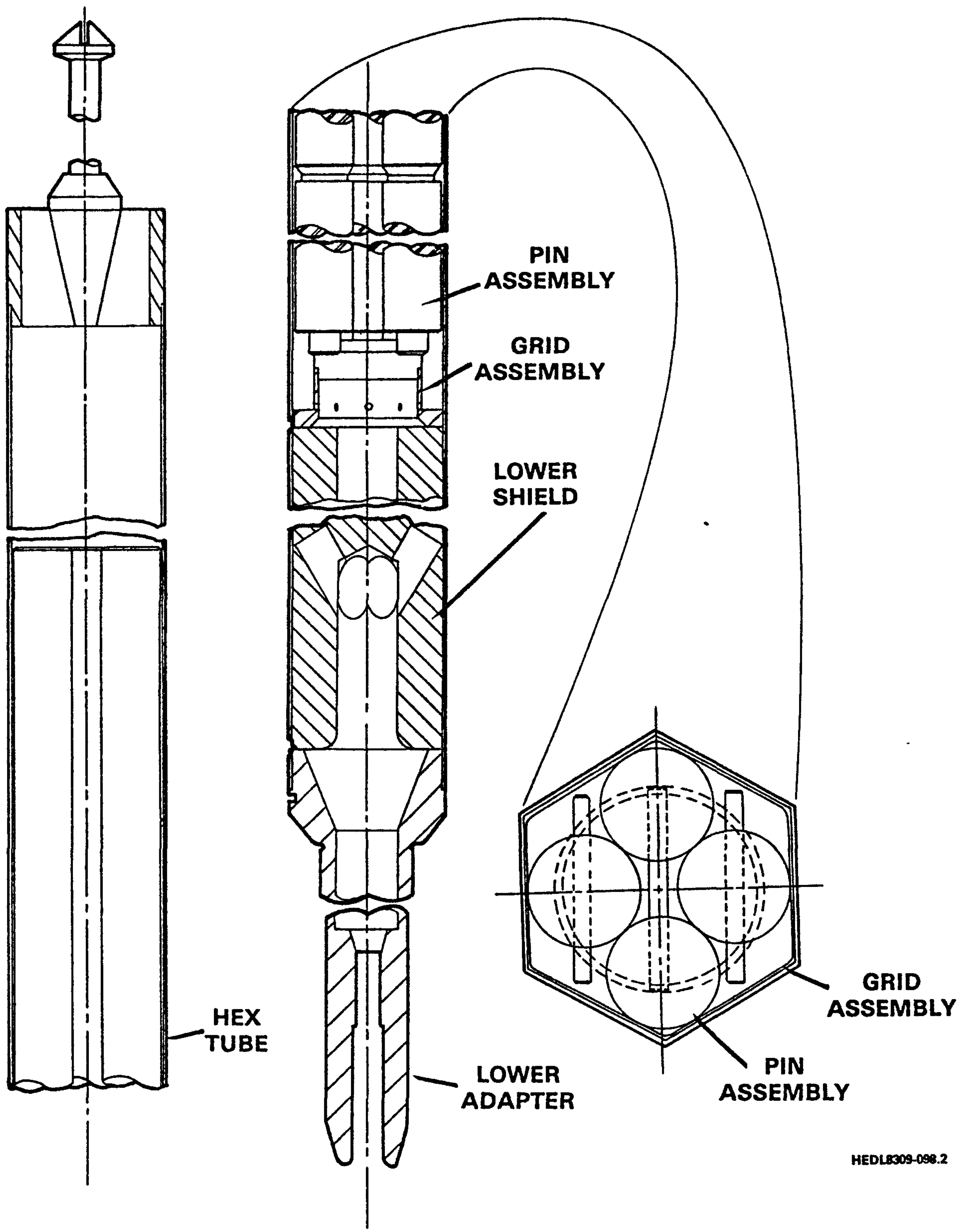


Fig. 7

SP-100 FUEL TEST CAPSULE

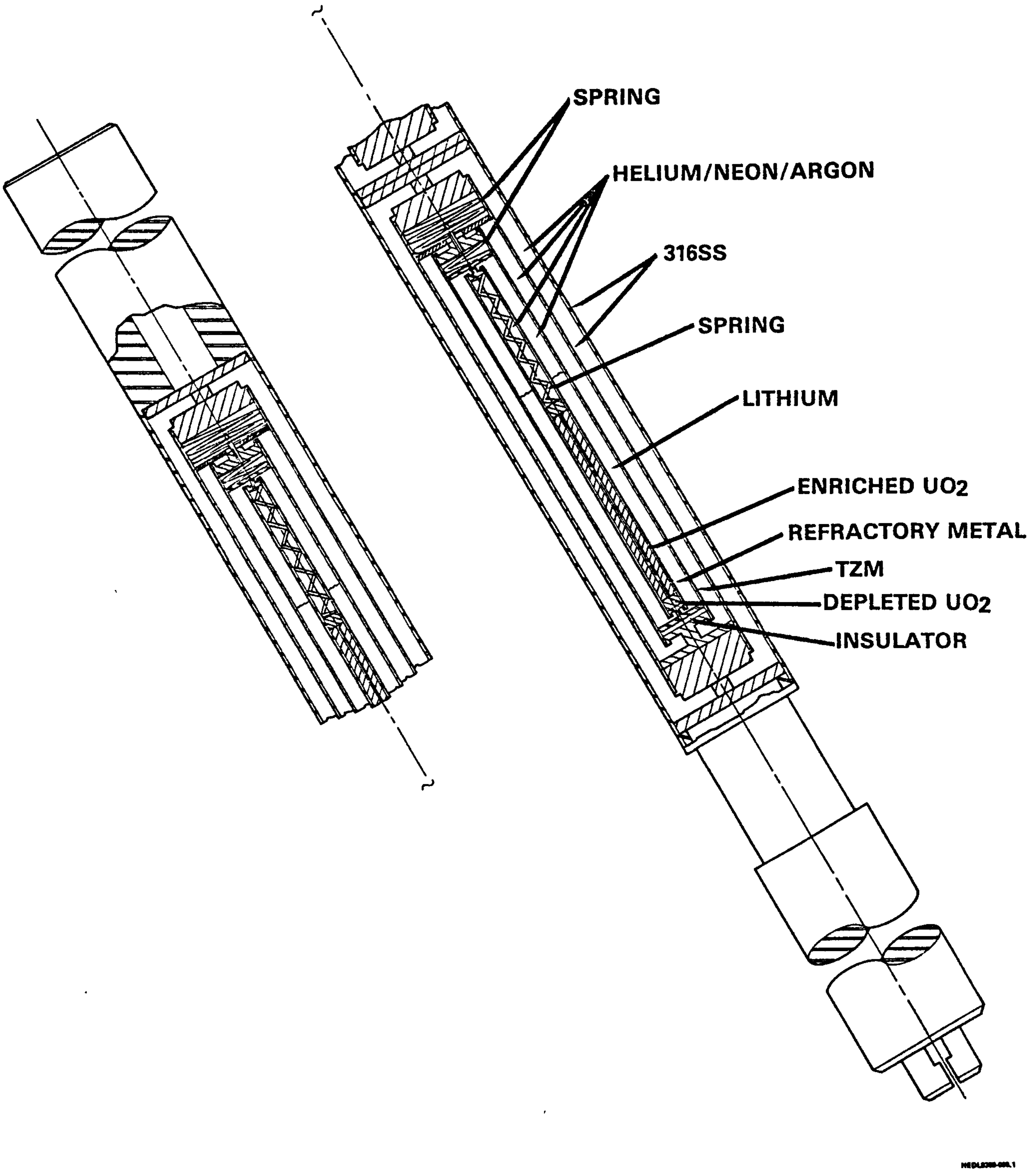


Fig. 8

CUTAWAY OF CAPSULES USED FOR EBR-II SPACE NUCLEAR POWER FUEL TEST

HEDL 8411-259.1

Fig. 9

SP-100 FUEL PIN FABRICATION PARAMETERS

	UN	UO_2	UO_2 HIGH DENSITY
PIN DIA., in.	0.30		
TYPE FUEL	SOLID PELLET	CORED PELLET	SOLID PELLET
PELLET DENSITY	95		
FUEL SMEAR DENSITY, %	80		86
PEAK CLADDING TEMP, K	1300, 1500		
CLADDING MAT'LS	W-26Re PWC-11 ASTAR-811C Nb-1Zr	Mo-13Re Nb-1Zr PWC-11	Nb-1Zr

HEDL 8410-052.4

Fig. 10

long fuel column. The fuel pin diameter was selected to provide a reasonable compromise between an accelerated test (accumulating burnup rapidly, which is accomplished by operating at a higher fission density than the SP-100 designs) and achieving prototypic environments. Peak cladding temperatures of 1300 K and 1500 K were chosen to address the SP-100 conditions. The fuel pin was sized to operate at a power density of 80 W/gm, approximately twice that expected in SP-100, yet to maintain prototypic fuel and cladding temperatures and heat flux's. This was accomplished with a smaller diameter pin (0.30 in. compared to a typical 0.50 in. SP-100 pin) and introducing a 0.070 in. central hole into the UO_2 fuel. The tests are scheduled to operate for 450 days at full power and will achieve prototypic levels of burnup ($\sim$5 at.%) and fast neutron fluence ($\sim 4.5 \times 10^{22}$ n/cm^2 >0.1 MeV). The fabrication parameters are shown in Figure 11. The enrichments for the standard UN fuel was chosen to assure the linear power of the UO_2 and UN fuel pins were the same. The enrichment for the high density solid pellet UO_2 fuel was chosen to keep the fuel temperatures of both UO_2 types the same.

The test matrix for the two assemblies is shown in Figures 12 and 13. At the time of the SP-1 interim examination one UO_2 and one UN capsule will be removed from the assembly and replaced with Astar-811C clad UN pins and PWC-11 clad UO_2 pins. A high density UO_2 capsule will be removed from SP-2 and replaced with Mo/Re clad UO_2 fuel pins.

A reasonably accurate estimate of the cladding surface temperatures is critical to the interpretation of test results. The calculation of temperature increases across the two insulating gas gaps is relatively straightforward. The calculation of heat transfer through the lithium and the pin end effects required more sophisticated procedures using a two dimensional heat transfer code. Figure 14 shows the results of parametric studies in which the effect of thermally induced lithium convection was investigated.

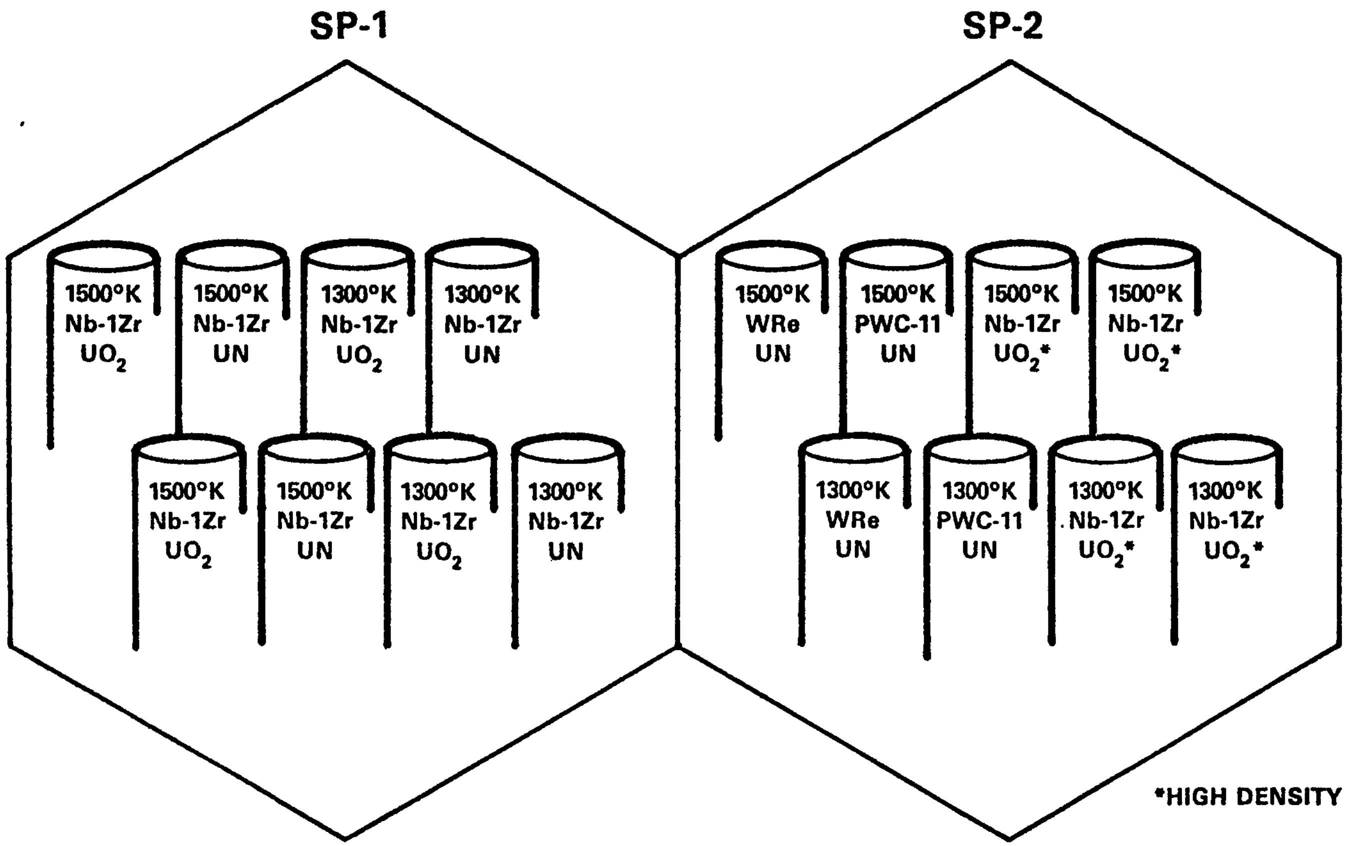
The low fuel smear density in the pin provides sufficient room to accommodate the anticipated fuel swelling. Consequently, the primary cladding loading

COMPARISON OF PARAMETERS
IN HIGH DENSITY AND LOW DENSITY
UO_2 FUEL

<table>
<tr><th></th><th>HIGH DENSITY</th><th>LOW DENSITY</th></tr>
<tr><td>PELLET DIAMETER, INCH</td><td colspan="2">.240</td></tr>
<tr><td>CENTRAL VOID, INCH</td><td>0.</td><td>.070</td></tr>
<tr><td>PELLET DENSITY, %TD</td><td colspan="2">95</td></tr>
<tr><td>SMEAR DENSITY, %TD</td><td>86</td><td>80</td></tr>
<tr><td>ENRICHMENT, % U235</td><td>35</td><td>47</td></tr>
<tr><td>LINEAR POWER, KW/FT</td><td>6.1</td><td>7.5</td></tr>
<tr><td>PREDICTED PEAK FUEL TEMPERATURE, K</td><td colspan="2">2420</td></tr>
<tr><td>FUEL COLUMN LENGTH, INCH</td><td>2</td><td>3</td></tr>
<tr><td>PLENUM TO FUEL–RATIO</td><td>1.5</td><td>.7</td></tr>
</table>

Fig. 11

CONFIGURATION OF THE SP-100 FUEL PIN IRRADIATION TESTS



HEDL 8410-052.2

Fig. 12

CONFIGURATION OF THE SP-100 FUEL PIN IRRADIATION TESTS AFTER RECONSTITUTION

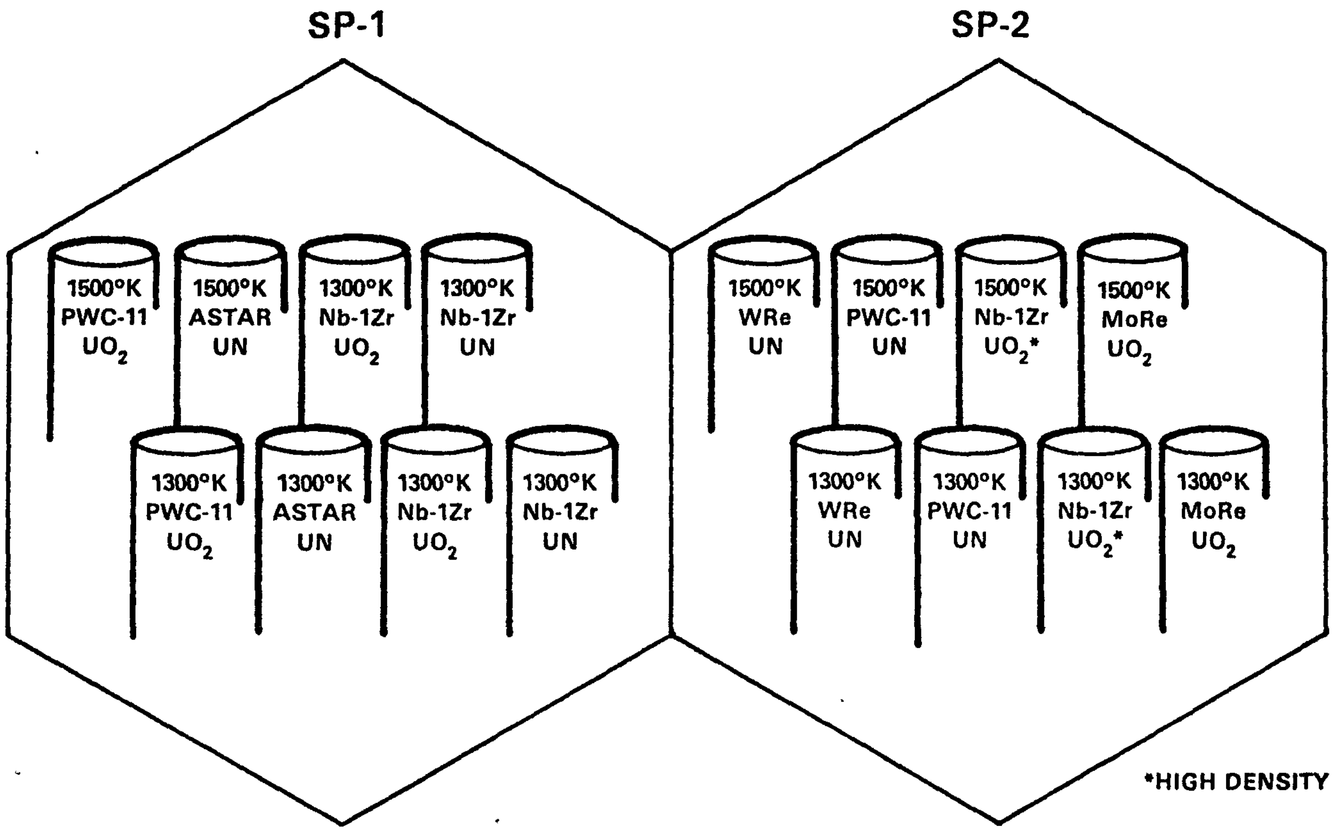


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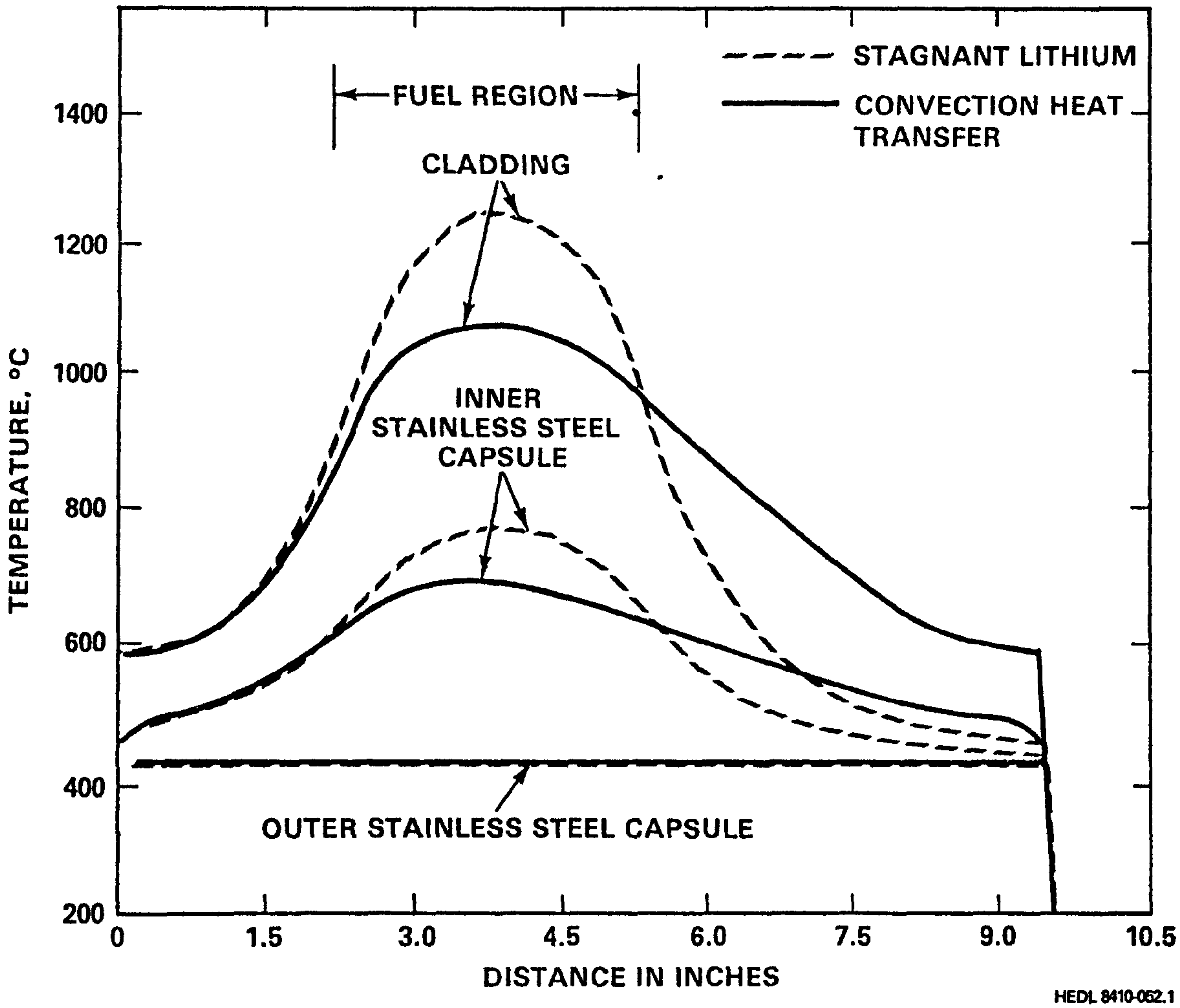
Fig. 13

EFFECT OF LITHIUM CONVECTION ON CAPSULE HEAT TRANSPORT



HEDL 8410-062.1

Fig. 14

is due to fission gas pressure. The test pin's performance was calculated using a modified version of the SIEX computer code (Dutt, Baker 1975) which was developed for fast reactor fuel element design.

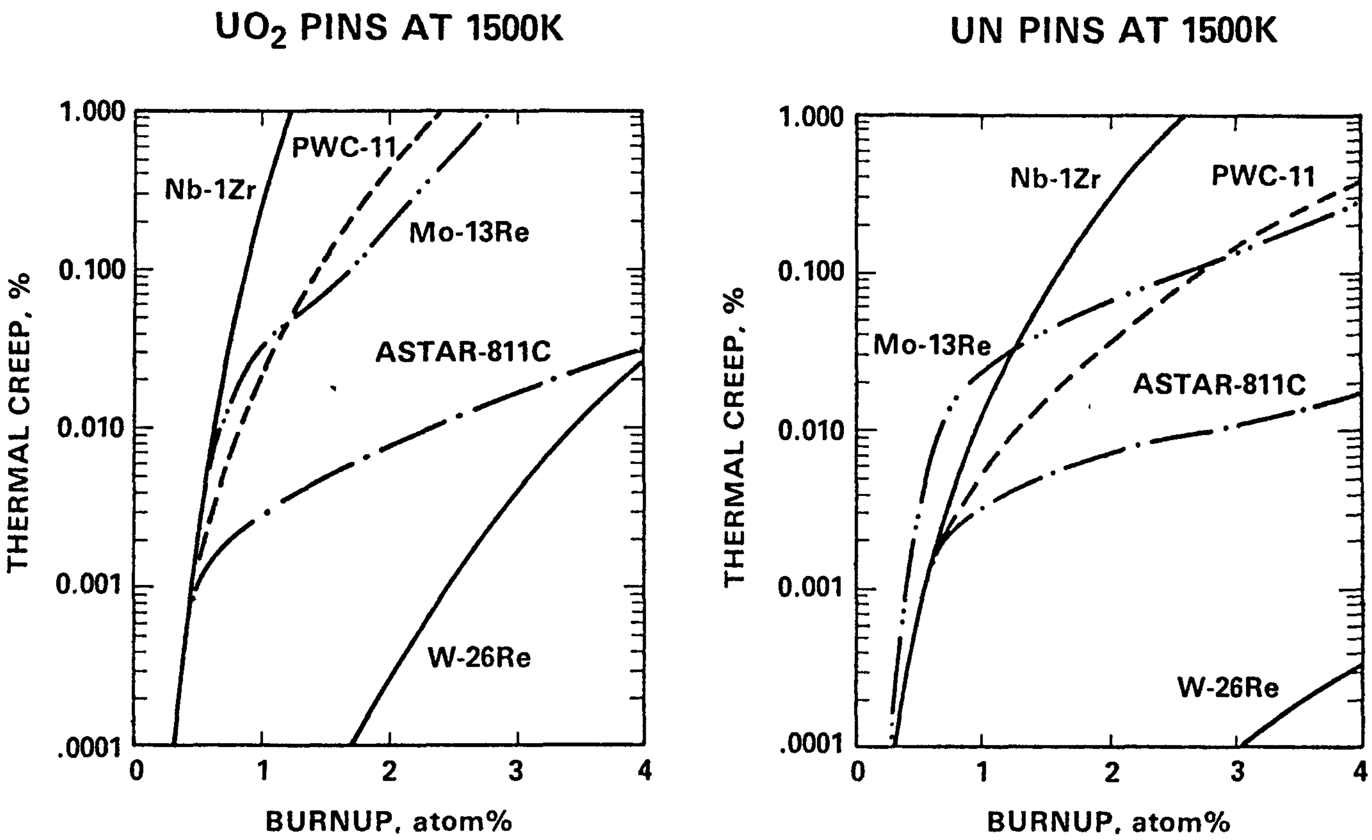
The performance models in this code are well documented and validated, providing confidence in the predicted fuel temperatures and UO_2 gas release.

The basis for gas release and thermal performance models for UN fuel in a fast neutron environment is not as well developed as that for UO_2. Rather than try to develop validated UN models for gas release, gap closure, etc., based on inconsistent data, a conservative value for gas release (20%) was used and the gap conductance model developed for UO_2 was retained. Of course, the UN conductivity was incorporated as an option in the code. It is believed these assumptions provide conservative predictions of fuel element performance consistent with that required for design and safety analyses.

Using these fuel performance models and recommended properties for refractory alloys, the calculated behavior of the test on pins is shown in Figure 15 and for the UO_2 test pins in Figures 16 and 17. All the fuel and cladding combinations except the high temperature Nb-1Zr clad pins will meet the 1% strain criterion usually used in design of refractory clad fuel elements. The high temperature Nb-1Zr clad pins will be removed after 1 at.% burnup for interim examination, they will not experience excessive strain and will provide an indication of fuel element performance under the most aggressive condition. These calculations and associated safety analyses show that the two tests will operate as planned to a burnup of 4 to 5 at.% without cladding breach.

The schedule for irradiation and examination of the tests is shown in Figure 18. The interim examination of SP-1 will be conducted at $\sim$1 at.% burnup to provide feedback by the summer of 1985. At the time of the interim examinations all fuel pins will be gamma scanned and neutron radiographed to determine whether failures occurred and to determine fuel length and diameter

CALCULATED THERMAL CREEP IN REFRACTORY ALLOY CLAD SP-1 AND SP-2 FUEL PINS



HEDL 8412-099

Fig. 15

CALCULATED THERMAL CREEP FOR UO2 FUEL PIN
OPERATING AT 1300°K CLAD TEMPERATURE
VERSUS
EBR-II IRRADIATION TIME

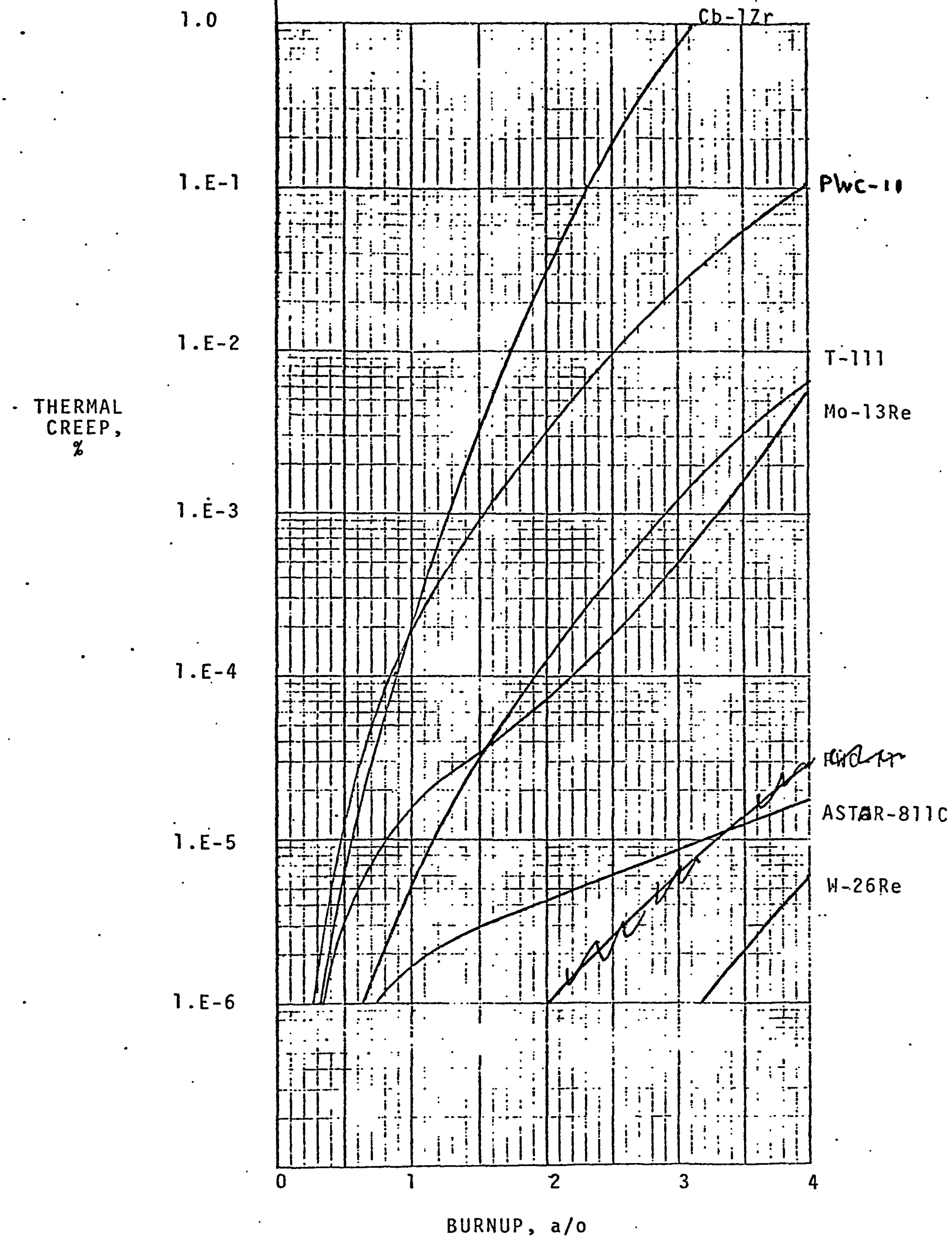


Fig. 16

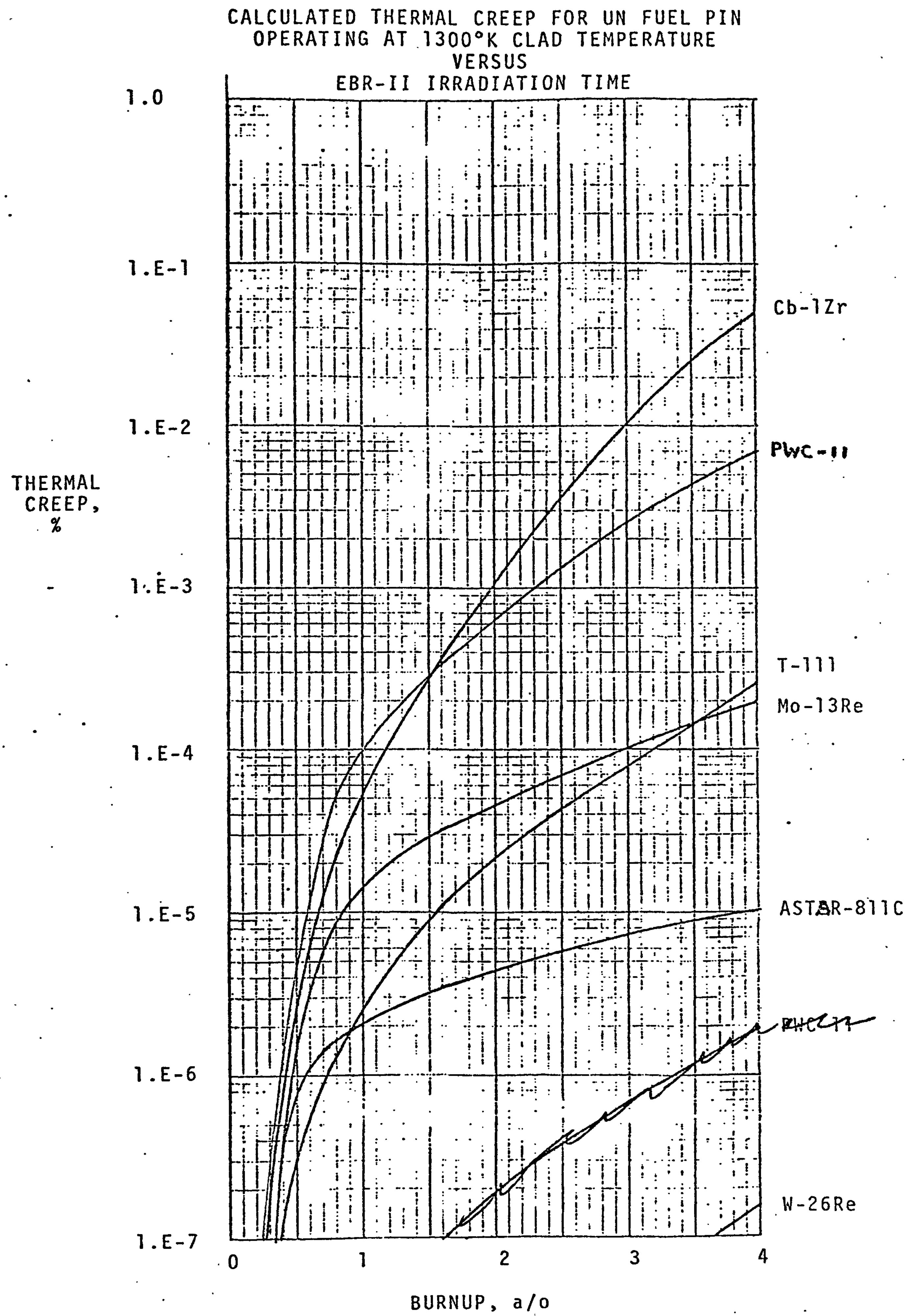

CALCULATED THERMAL CREEP FOR UN FUEL PIN
OPERATING AT 1300°K CLAD TEMPERATURE
VERSUS
EBR-II IRRADIATION TIME
THERMAL CREEP, %
1.0
1.E-1
1.E-2
1.E-3
1.E-4
1.E-5
1.E-6
1.E-7
0
1
2
3
4
BURNUP, a/o
Cb-1Zr
PWC-11
T-111
Mo-13Re
ASTAR-811C
W-26Re

Fig. 17

SP-100 FABRICATION, IRRADIATION, AND EXAMINATION SCHEDULE FOR THE FUEL PIN IRRADIATION TESTS

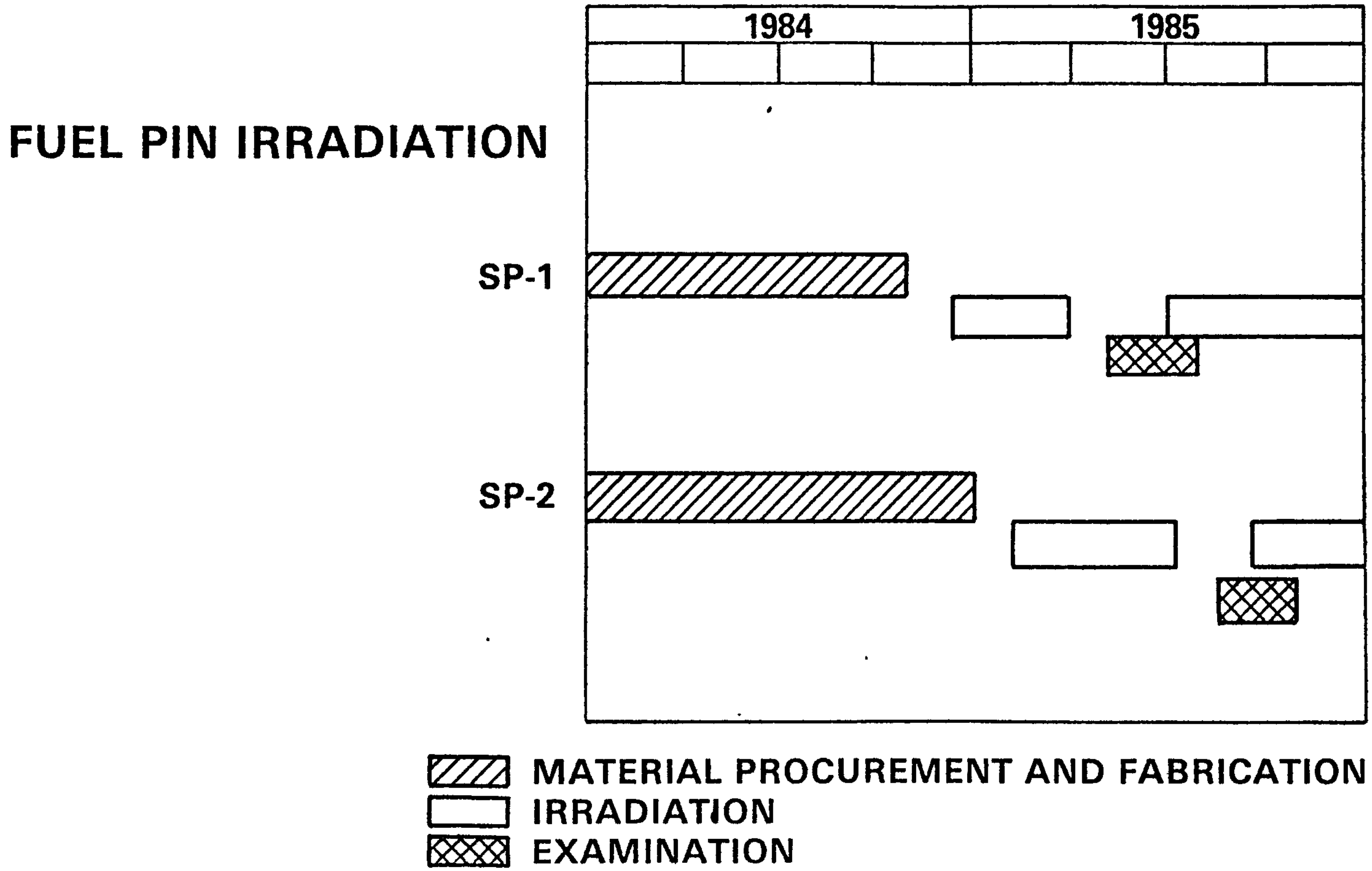


Fig. 18

HEDL 8410-088

changes ($\pm$1 mil). Two capsules (four fuel pins) will be removed from the test assembly at this time for destructive examination to evaluate fuel/cladding reactions and to obtain benchmark data on fuel swelling and fission gas release. The assembly will be reconstituted with new fuel pins and continue irradiation to goal burnup in early 1987. On reaching goal burnup all pins will be nondestructively and destructively examined for dimensional changes, chemical reactions, fission gas release, and fuel swelling.

This work was performed at the Hanford Engineering Development Laboratory, which is operated for the Department of Energy by Westinghouse Hanford Company under Contract No. DE-AC06-76FF02170.

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EDB	X	INS	X	TTR	☐	H	☐	1. ~~220800~~
ERA	X	RIP	☐	DOE	X	SPF	☐	2. 210600
EPA	☐	ATF	☐	LRF	☐	LLF	☐	3.
WDA	☐	CRF	☐	SNI	☐	LTF	☐	4.
CRA	☐	WDF	☐	HGF	☐		☐	5.
								6.

AUGMENTATION:

DESCRIPTORS	FLAGS
Space ~~Propulsion~~ Power Reactors	T1
Fuel Pins	Q1
Irradiation	
Physical Radiation Effects	
Burnup	
Fuel Cans	T2, Q1
Very High Temperature	
Computer Calculations	
Materials Testing	Q2
Heat Transfer	
~~Testing~~	
Fabrication	
Reactor Materials	

DH / JWB

FORM TI-70 (7-85)

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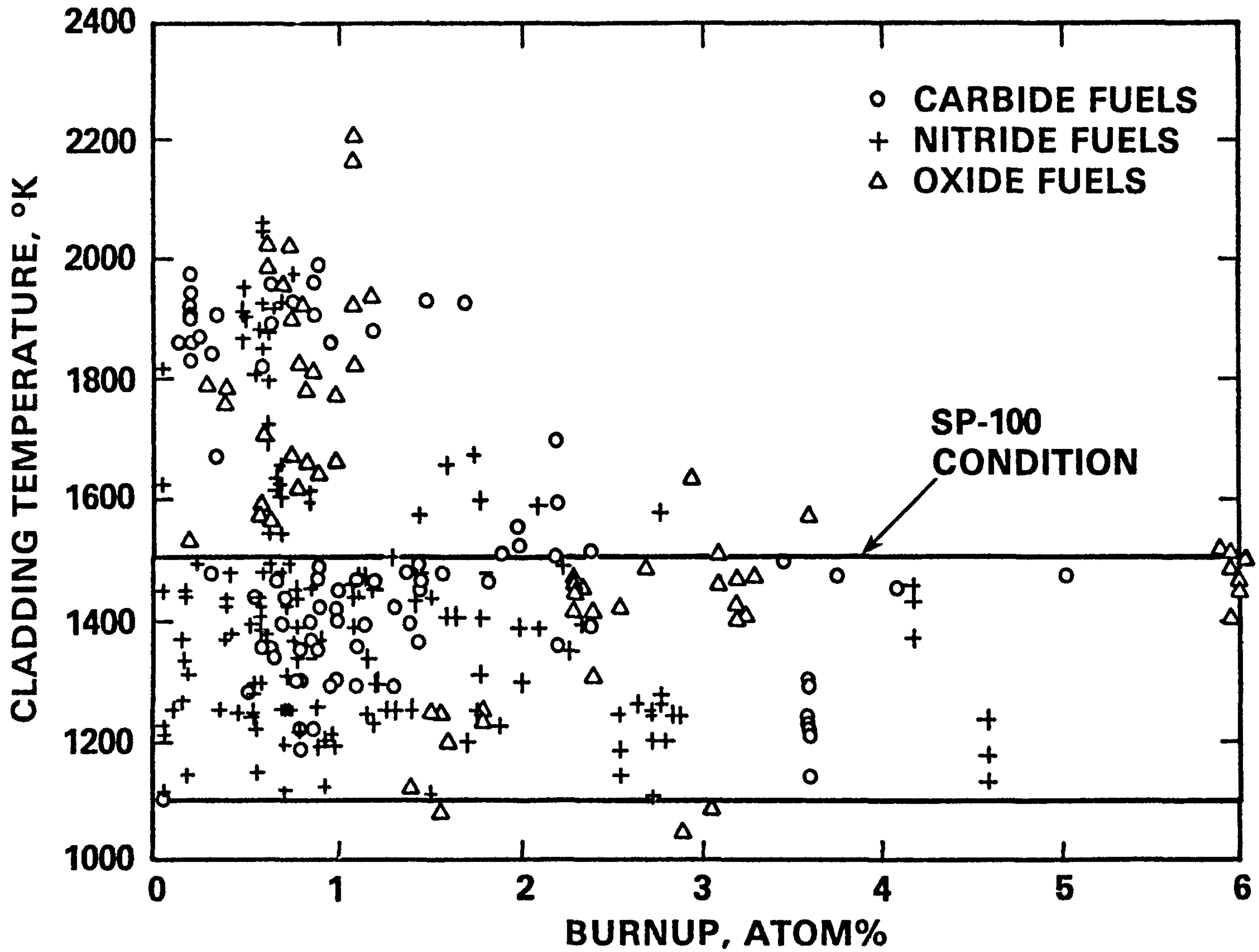

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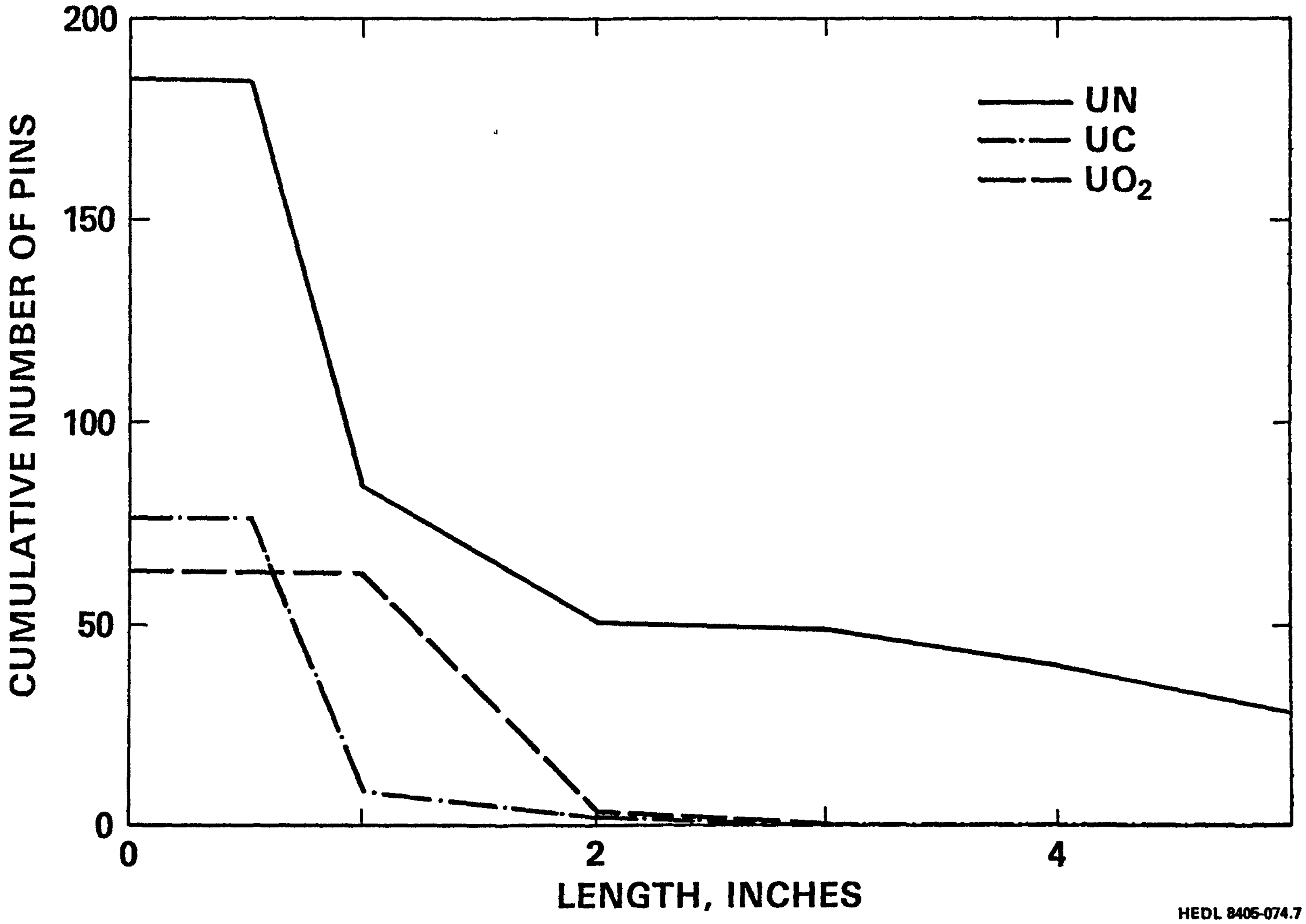
Fig. 4

IRRADIATION DATA BASE

RANGE OF LENGTHS



FUEL AND CLADDING CANDIDATES FOR SP-100 APPLICATION

FUELS	CLADDINGS
UN	W-26Re
	Ta-8W-1Re-0.7Hf-0.025C (ASTAR-811C)
	Mo-13Re
UO_2	Nb-1Zr
	Nb-1Zr-0.1C (PWC-11)

Fig. 5

OUTSTANDING ISSUES

- HIGH BURNUP SWELLING OF UN
- FUEL/CLADDING COMPATIBILITY
- DEVELOPMENT OF DATA BASE WITH PROTOTYPICALLY CONFIGURED FUEL PINS AND FAST NEUTRON IRRADIATION
- EFFECTS OF UO_2-Li REACTION IN EVENT OF PIN FAILURE

Fig. 6

These issues must be addressed by a contemporary testing program to assure the feasibility of such fuel pins under the high temperature, fast neutron flux, and long lifetime conditions of SP-100.

SP-1 and SP-2 have been designed and partially constructed for irradiation in the Experimental Breeder Reactor-II (EBR-II) to address the first three feasibility issues. SP-1 will provide interim examination data to assist in the fuel system selection in July 1985, and both experiments will be completed, to SP-100 goal burnup, by 1987. Irradiation of SP-1 began in December 1984 with fabrication of SP-2 proceeding towards a mid-1985 beginning of irradiation.

The irradiation test vehicle selected for the experiment is a modified version of a proven irradiation test vehicle used for high temperature B_4C irradiations for the LMFBR program. The test vehicle contains four test subassemblies, Figure 7. The outer surfaces of these test subassemblies, Figure 8, are exposed to the reactor sodium coolant at $\sim$650 K. A stainless steel subcapsule is used to increase the temperature to the desired fuel pin cladding temperature, 1300 to 1500 K, by providing two insulating gas gaps Figure 9. The innermost capsule, fabricated from TZM, a molybdenum alloy, is used to contain each fuel pin submerged in lithium. The lower TZM capsule in each assembly is designed so a thermal expansion device, TED, can be included to assist, via postirradiation examinations, in determining operating temperatures. Preirradiation design uncertainties in operating temperature, including the effects of natural circulation in the lithium filled annulus are $\sim\pm$60 K. Postirradiation measurements of the TED's coupled with burnup analyses are expected to reduce this to $\sim\pm$30 K.

Fuel pin fabrication parameters for the experiments, Figure 10, was influenced by the desire to obtain nearly prototypic, but accelerated (relative to the SP-100 seven year lifetime) data, and the constraints of EBR-II. The capsules were designed to accommodate two fuel pins in each of the four test positions; making a total of eight fuel pins in each test vehicle. To fit within this envelope the test fuel pins are 6.4 in. long and contain a 3 in.

SP-100 FUEL TEST ASSEMBLY

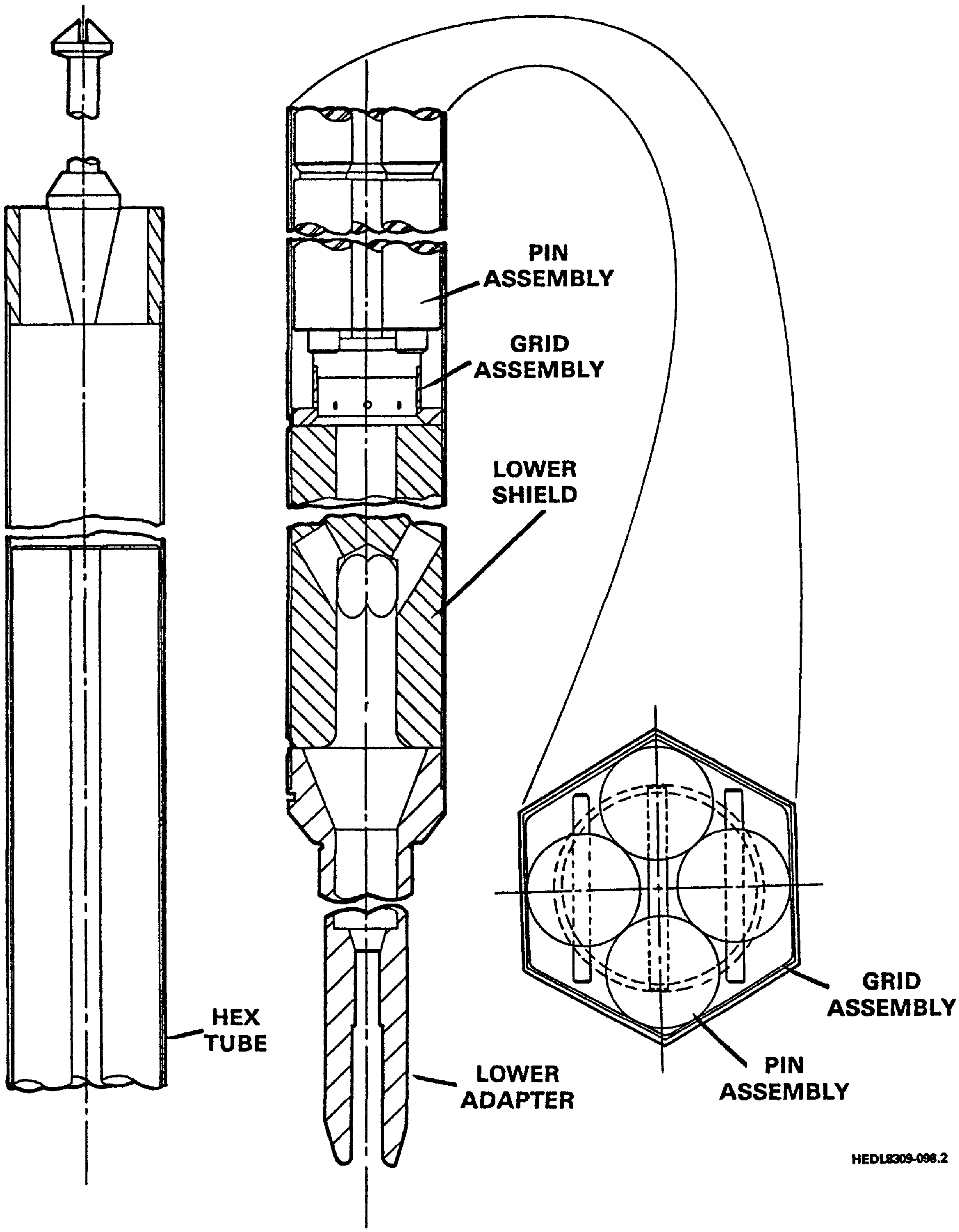


Fig. 7

SP-100 FUEL TEST CAPSULE

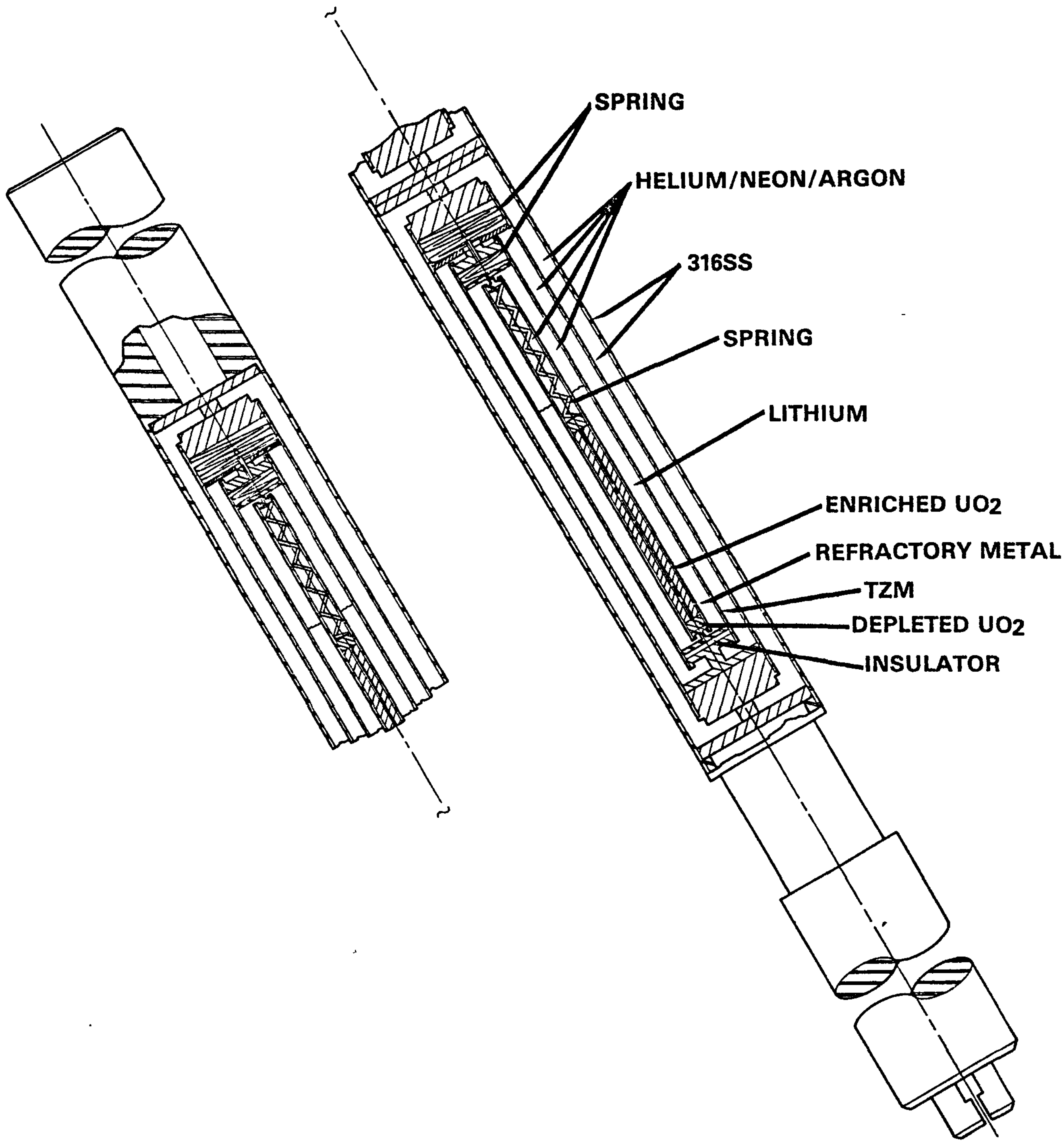


Fig. 8

CUTAWAY OF CAPSULES USED FOR EBR-II SPACE NUCLEAR POWER FUEL TEST

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Fig. 9

SP-100 FUEL PIN FABRICATION PARAMETERS

	UN	UO_2	UO_2 HIGH DENSITY
PIN DIA., in.	0.30		
TYPE FUEL	SOLID PELLET	CORED PELLET	SOLID PELLET
PELLET DENSITY	95		
FUEL SMEAR DENSITY, %	80		86
PEAK CLADDING TEMP, K	1300, 1500		
CLADDING MAT'LS	W-26Re PWC-11 ASTAR-811C Nb-1Zr	Mo-13Re Nb-1Zr PWC-11	Nb-1Zr

HEDL 8410-052.4

Fig. 10

long fuel column. The fuel pin diameter was selected to provide a reasonable compromise between an accelerated test (accumulating burnup rapidly, which is accomplished by operating at a higher fission density than the SP-100 designs) and achieving prototypic environments. Peak cladding temperatures of 1300 K and 1500 K were chosen to address the SP-100 conditions. The fuel pin was sized to operate at a power density of 80 W/gm, approximately twice that expected in SP-100, yet to maintain prototypic fuel and cladding temperatures and heat flux's. This was accomplished with a smaller diameter pin (0.30 in. compared to a typical 0.50 in. SP-100 pin) and introducing a 0.070 in. central hole into the UO_2 fuel. The tests are scheduled to operate for 450 days at full power and will achieve prototypic levels of burnup ($\sim$5 at.%) and fast neutron fluence ($\sim 4.5 \times 10^{22}$ n/cm^2 >0.1 MeV). The fabrication parameters are shown in Figure 11. The enrichments for the standard UN fuel was chosen to assure the linear power of the UO_2 and UN fuel pins were the same. The enrichment for the high density solid pellet UO_2 fuel was chosen to keep the fuel temperatures of both UO_2 types the same.

The test matrix for the two assemblies is shown in Figures 12 and 13. At the time of the SP-1 interim examination one UO_2 and one UN capsule will be removed from the assembly and replaced with Astar-811C clad UN pins and PWC-11 clad UO_2 pins. A high density UO_2 capsule will be removed from SP-2 and replaced with Mo/Re clad UO_2 fuel pins.

A reasonably accurate estimate of the cladding surface temperatures is critical to the interpretation of test results. The calculation of temperature increases across the two insulating gas gaps is relatively straightforward. The calculation of heat transfer through the lithium and the pin end effects required more sophisticated procedures using a two dimensional heat transfer code. Figure 14 shows the results of parametric studies in which the effect of thermally induced lithium convection was investigated.

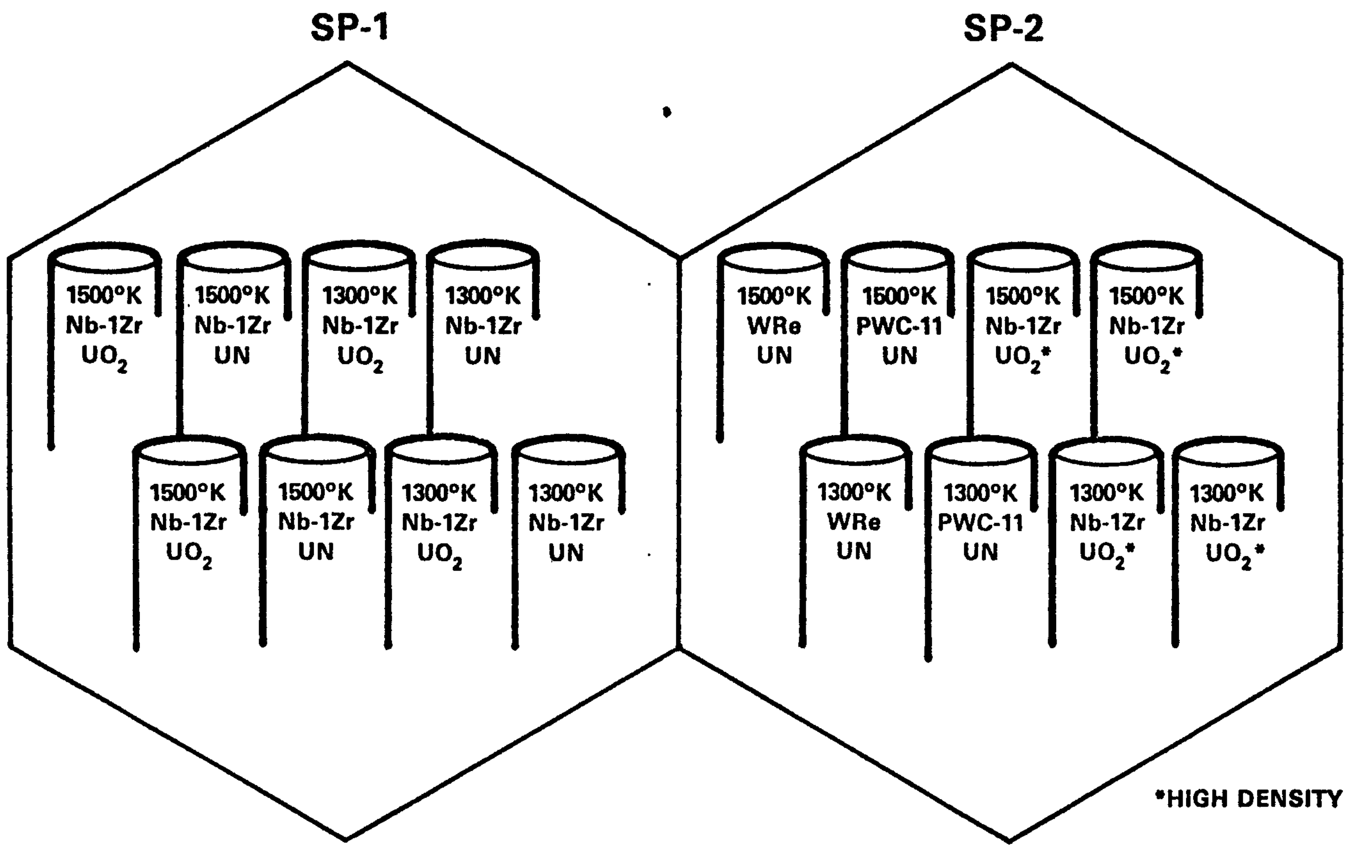
The low fuel smear density in the pin provides sufficient room to accommodate the anticipated fuel swelling. Consequently, the primary cladding loading

COMPARISON OF PARAMETERS
IN HIGH DENSITY AND LOW DENSITY
UO_2 FUEL

<table>
<tr><th></th><th>HIGH DENSITY</th><th>LOW DENSITY</th></tr>
<tr><td>PELLET DIAMETER, INCH</td><td colspan="2">.240</td></tr>
<tr><td>CENTRAL VOID, INCH</td><td>0.</td><td>.070</td></tr>
<tr><td>PELLET DENSITY, %TD</td><td colspan="2">95</td></tr>
<tr><td>SMEAR DENSITY, %TD</td><td>86</td><td>80</td></tr>
<tr><td>ENRICHMENT, % U235</td><td>35</td><td>47</td></tr>
<tr><td>LINEAR POWER, KW/FT</td><td>6.1</td><td>7.5</td></tr>
<tr><td>PREDICTED PEAK FUEL TEMPERATURE, K</td><td colspan="2">2420</td></tr>
<tr><td>FUEL COLUMN LENGTH, INCH</td><td>2</td><td>3</td></tr>
<tr><td>PLENUM TO FUEL-RATIO</td><td>1.5</td><td>.7</td></tr>
</table>

Fig. 11

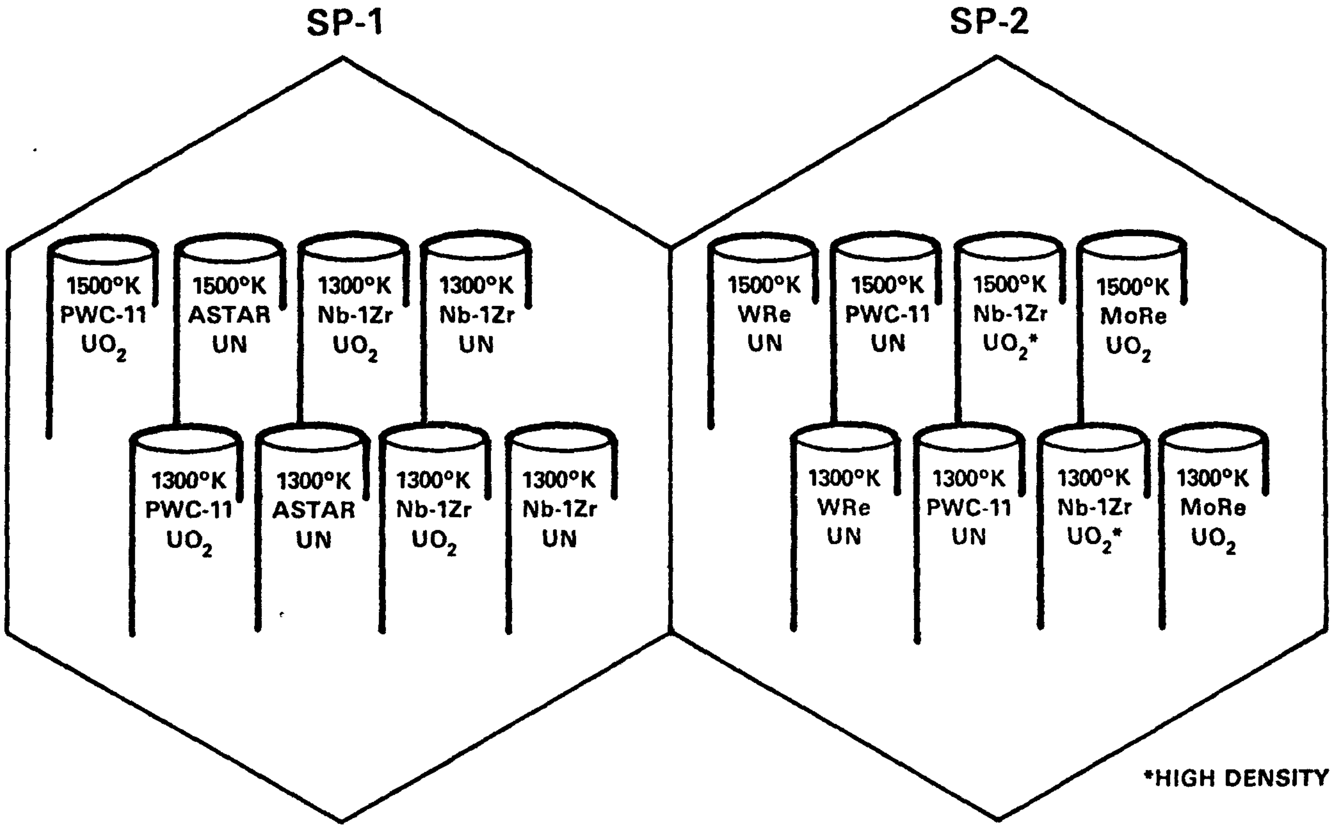
CONFIGURATION OF THE SP-100 FUEL PIN IRRADIATION TESTS



HEDL 8410-052.2

Fig. 12

CONFIGURATION OF THE SP-100 FUEL PIN IRRADIATION TESTS AFTER RECONSTITUTION



HEDL 8410-182

Fig. 13

EFFECT OF LITHIUM CONVECTION ON CAPSULE HEAT TRANSPORT

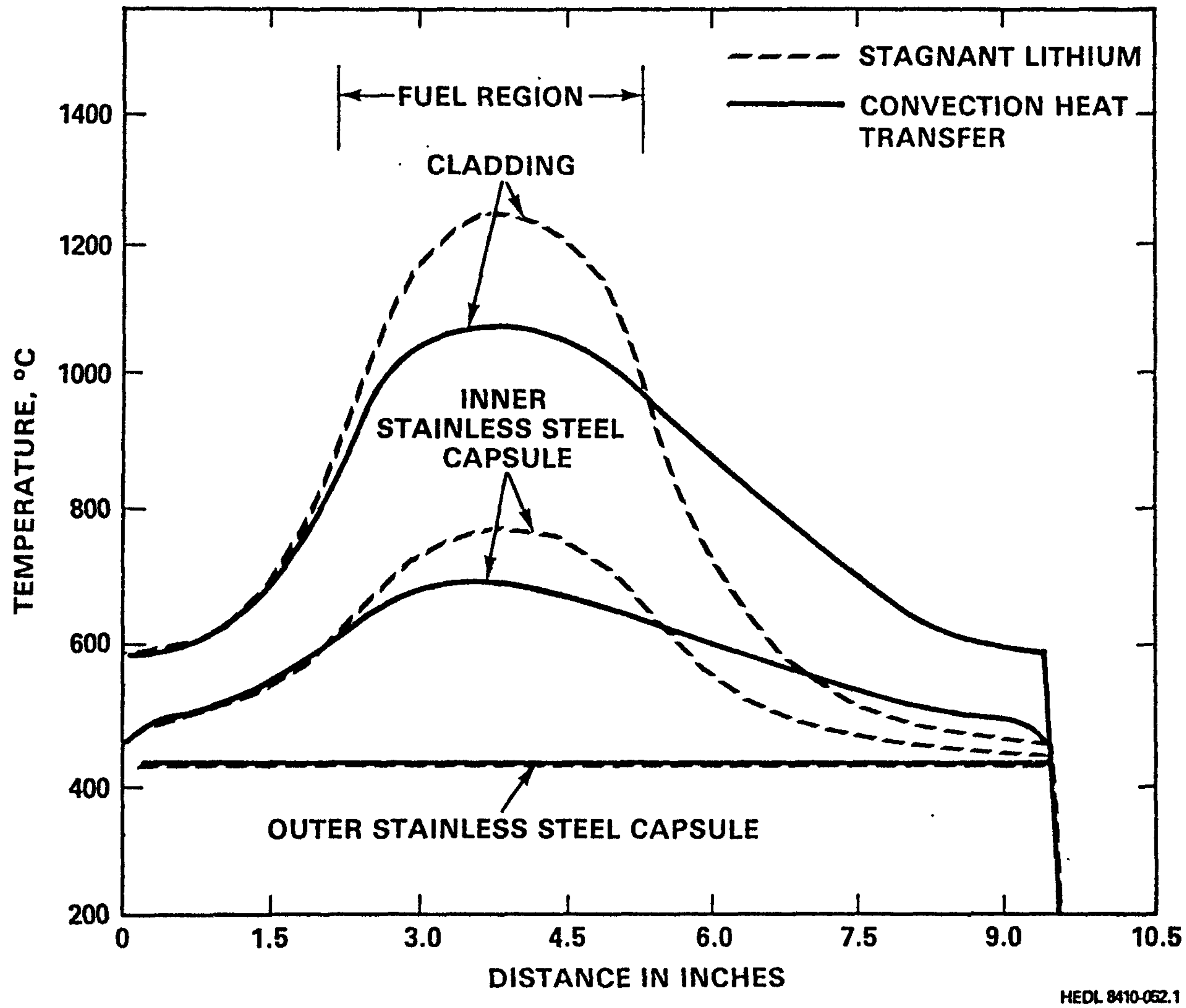


Fig. 14

is due to fission gas pressure. The test pin's performance was calculated using a modified version of the SIEX computer code (Dutt, Baker 1975) which was developed for fast reactor fuel element design.

The performance models in this code are well documented and validated, providing confidence in the predicted fuel temperatures and UO_2 gas release.

The basis for gas release and thermal performance models for UN fuel in a fast neutron environment is not as well developed as that for UO_2. Rather than try to develop validated UN models for gas release, gap closure, etc., based on inconsistent data, a conservative value for gas release (20%) was used and the gap conductance model developed for UO_2 was retained. Of course, the UN conductivity was incorporated as an option in the code. It is believed these assumptions provide conservative predictions of fuel element performance consistent with that required for design and safety analyses.

Using these fuel performance models and recommended properties for refractory alloys, the calculated behavior of the test on pins is shown in Figure 15 and for the UO_2 test pins in Figures 16 and 17. All the fuel and cladding combinations except the high temperature Nb-1Zr clad pins will meet the 1% strain criterion usually used in design of refractory clad fuel elements. The high temperature Nb-1Zr clad pins will be removed after 1 at.% burnup for interim examination, they will not experience excessive strain and will provide an indication of fuel element performance under the most aggressive condition. These calculations and associated safety analyses show that the two tests will operate as planned to a burnup of 4 to 5 at.% without cladding breach.

The schedule for irradiation and examination of the tests is shown in Figure 18. The interim examination of SP-1 will be conducted at $\sim$1 at.% burnup to provide feedback by the summer of 1985. At the time of the interim examinations all fuel pins will be gamma scanned and neutron radiographed to determine whether failures occurred and to determine fuel length and diameter

CALCULATED THERMAL CREEP IN REFRACTORY ALLOY CLAD SP-1 AND SP-2 FUEL PINS

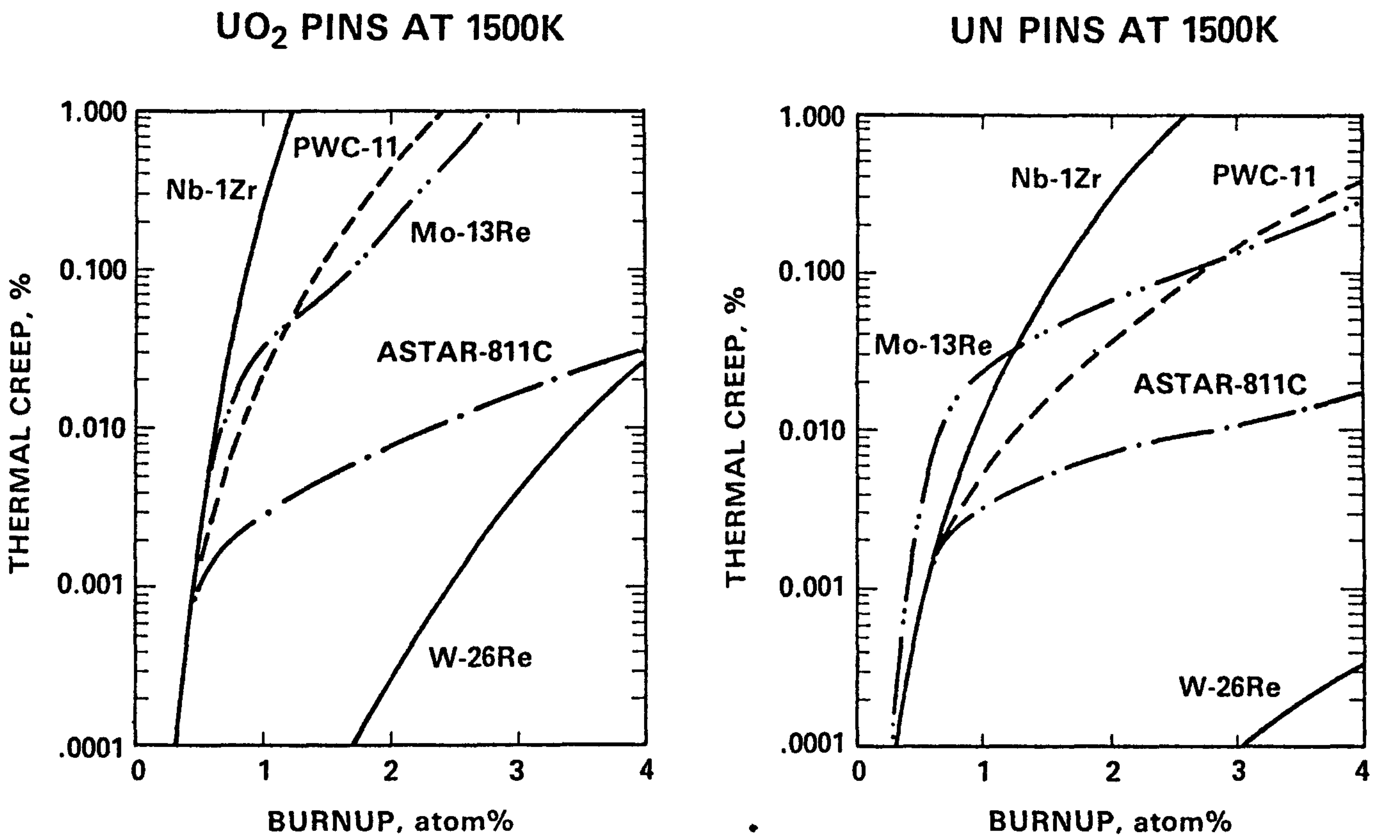


Fig. 15

HEDL 8412-099

CALCULATED THERMAL CREEP FOR UO2 FUEL PIN
OPERATING AT 1300°K CLAD TEMPERATURE
VERSUS
EBR-II IRRADIATION TIME

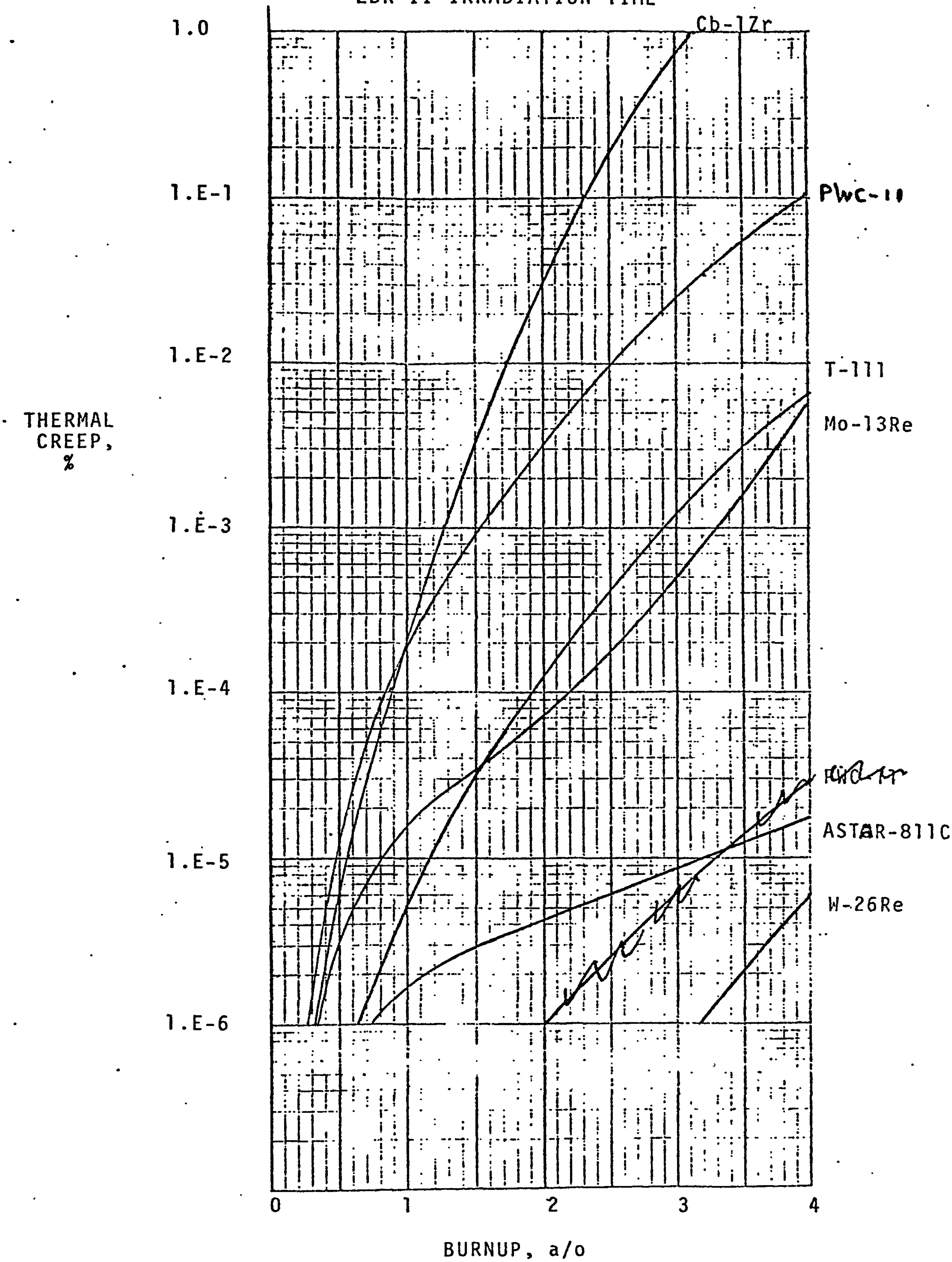


Fig. 16

CALCULATED THERMAL CREEP FOR UN FUEL PIN OPERATING AT 1300°K CLAD TEMPERATURE VERSUS EBR-II IRRADIATION TIME

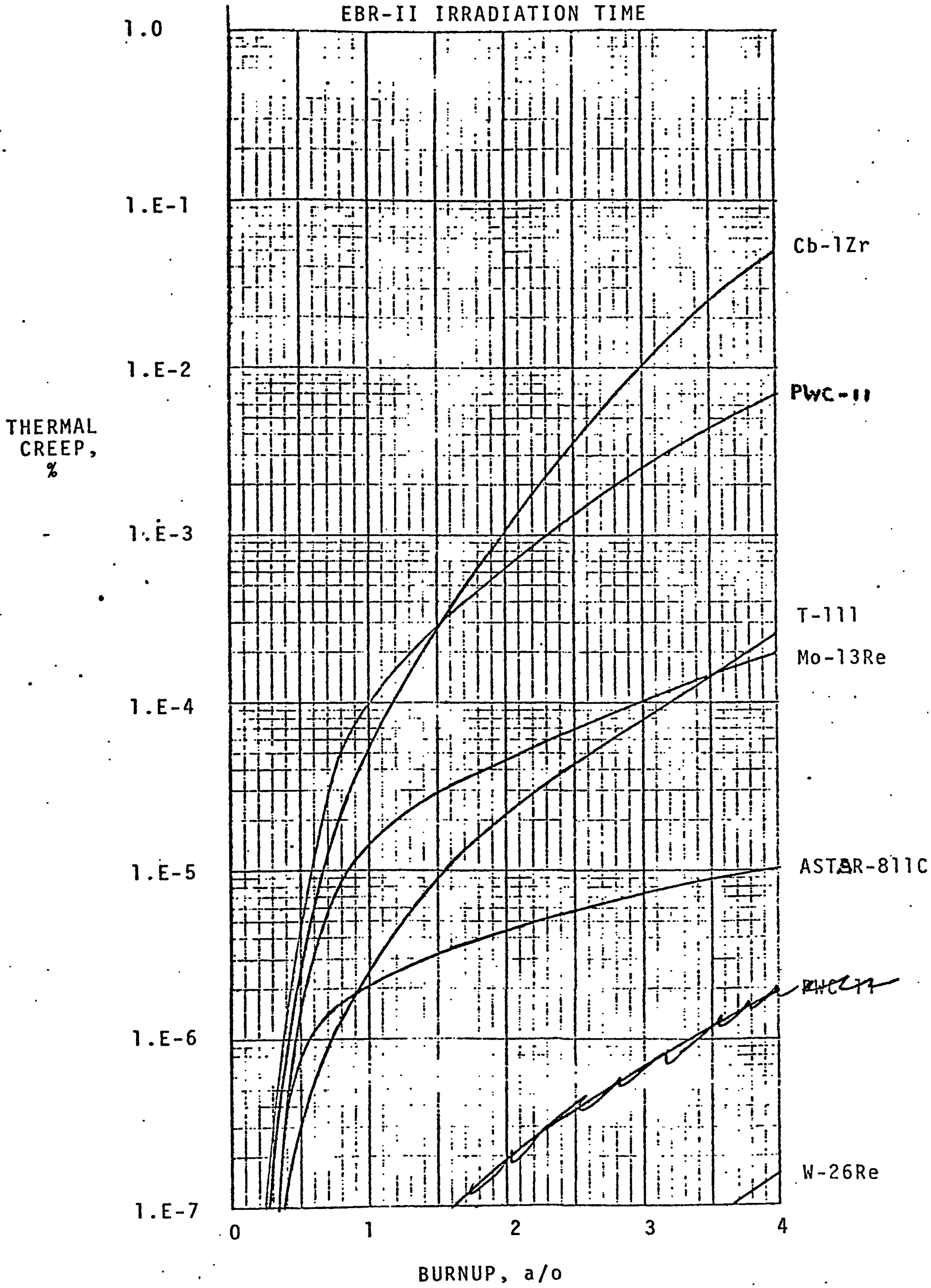
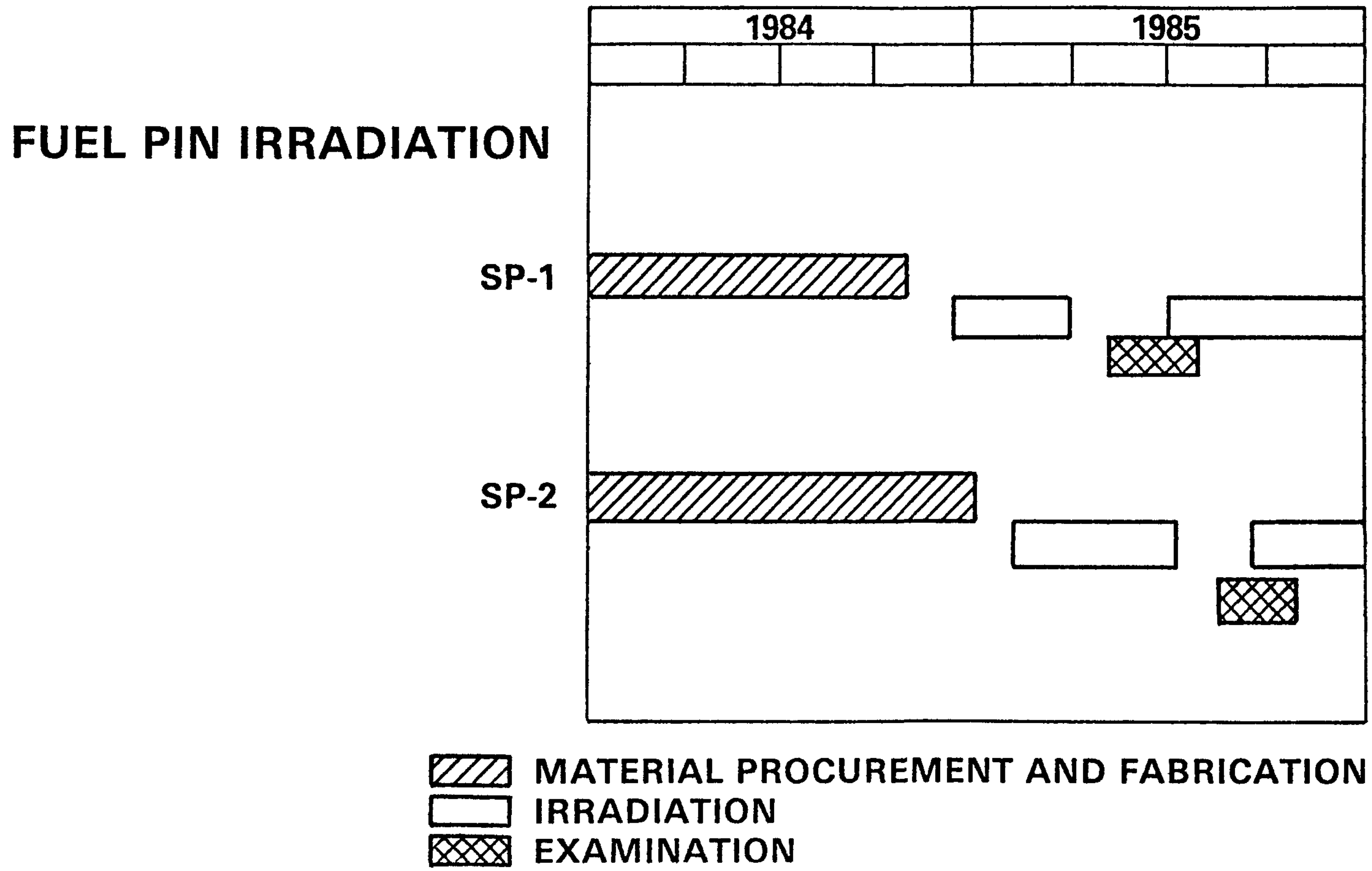


Fig. 17

SP-100 FABRICATION, IRRADIATION, AND EXAMINATION SCHEDULE FOR THE FUEL PIN IRRADIATION TESTS



HEDL 8410-088

Fig. 18

changes ($\pm$1 mil). Two capsules (four fuel pins) will be removed from the test assembly at this time for destructive examination to evaluate fuel/cladding reactions and to obtain benchmark data on fuel swelling and fission gas release. The assembly will be reconstituted with new fuel pins and continue irradiation to goal burnup in early 1987. On reaching goal burnup all pins will be nondestructively and destructively examined for dimensional changes, chemical reactions, fission gas release, and fuel swelling.

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