DOE-HTGR-86-025
Revision 2

HTGR

DESIGN DATA NEEDS MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR

MASTER

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ISSUED BY GA TECHNOLOGIES INC.
FOR THE DEPARTMENT OF ENERGY
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DOE–HTGR–86–025
Revision 2
PC–000216/3

DESIGN DATA NEEDS
MODULAR HIGH–TEMPERATURE
GAS–COOLED REACTOR

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PROJECT CONTROL DOCUMENT APPROVAL SUMMARY

	PROJECT	CODE	DOCUMENT NO.	REV.
DESIGN DATA NEEDS MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR	6300	DDN	PC-000216	3

C.M. STAMP	REV.	PREPARED BY	RESOURCE/ SUPPORT APPROVAL	FUNDING PROJECT APPROVAL	APPLICABLE PROJECT APPROVAL	DESCRIPTION/ CWBS NO.
REC'D CM MAR 07 1986	0	Project Staff			J.Wistrom 3.3.86. G.Bramblett 3-4-86	Initial Issue/ 6351010105
REC'D CM JUL 28 1986	1	Project Staff			J.Wistrom for G.Bramblett	6351010105 1. Revised DDNs 2. Added Safety significance and Quality Level.HTGR-86-025/1
REC'D CM SEP 19 1986	2	Project Staff			J.Wistrom G.Bramblett	6351 010 105 Added four more DDNs: M.10.18.21 thru 24
REC'D CM MAR 27 [illegible]	3	Project Staff			J.Wistrom for G.Bramblett	DOE-HTGR-86-025,Rev.2 6351 010 105 1. DDN's updated with RTDP, FAW's, schedule, priority and cost.

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Title Page	1	(Unnumbered)
Approval Summary	1	
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Total	392	

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TALABAY-35-A

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DESIGN DATA NEEDS

1. INTRODUCTION

The Design Data Needs (DDNs) provide summary statements for program management, of the designer's need for experimental data to confirm or validate assumptions made in the design. These assumptions were developed using the Integrated Approach and are tabulated in the Functional Analysis Report (Ref. 1). These assumptions were also necessary in the analyses or trade studies (A/TS) to develop selections of hardware design or design requirements. Each DDN includes statements providing traceability to the function and the associated assumption that requires the need.

2. DISCUSSION

The explicit requirements for securing experimental data are documented in:

A. Section 4 of the Technology Development plans for:

1. Graphite (Ref. 2).
2. Fuel and Fission Products, (Ref. 3).
3. Metals (Ref. 4).

B. Regulatory Technology Development Plan (RTDP) (Ref. 5)

C. Test Requirements Specifications for components not in the scope of (A) and (B) above.

A brief summary of these explicit requirements are given in the DDNs. In order to facilitate usage, the compilation of DDNs that follow is categorized and coded as shown in Table 1.

TABLE 1
HTGR DESIGN DATA NEEDS
IDENTIFICATION CODE

I. BY EFFECTIVITY

M) Modular High Temperature Gas-Cooled Reactor

II. BY APPLICABILITY

Designates Multiple Systems

M.01	Plant Performance
M.02	Availability and Maintenance
M.03	In-Service Inspection (ISI)
M.04	Plant Dynamics
M.05	Safety and Reliability
M.06	Plant Seismic
M.07	Fuel/Fission Product
M.08	Decay Heat Removal

Designates System/Subsystem		System/Subsystem
M.10	Reactor System	10
M.10.12	Neutron Control	10-12
M.10.17	Reactor Internals	10-17
M.10.18	Reactor Core	10-18
M.11	Vessel System	11
M.11.05	Pressure Relief	11-5
M.11.06	Vessel and Ducts	11-6
M.11.07	Vessel Support	11-7
M.20	Reactor Services Group	20
M.20.01	Hot Service Facility	20-1
M.20.02	Decontamination Services	20-2
M.20.16	Reactor Service Equip. & Stor. Wells	20-16
M.20.23	Helium Purification	20-23
M.20.24	Helium Storage and Trans.	20-24
M.20.25	Liquid Nitrogen	20-25
M.20.42	Essential Cooling Water	20-42
M.20.47	Reactor Plant Cooling Water	20-47
M.20.62	Liquid Radioactive Waste	20-62
M.20.64	Gaseous Radioactive Waste	20-64
M.20.65	Solid Radioactive Waste	20-65
M.21	Heat Transport System	21
M.21.01	Main Circulator	21-1
M.21.02	Steam Generator	21-2
M.30	Misc. Control & Inst. Group	30
M.30.01	NSSS Analytical Instrumentation System	30-1
M.30.03	Radiation Monitoring	30-3
M.30.04	Seismic Monitoring	30-4
M.30.05	Meterorogical Monitoring	30-5
M.30.06	Fire Detection and Alarm	30-6

TABLE 1 (Cont'd.)

Designates System/Subsystem		System/Subsystem
M.30.07	Security Monitoring	30-7
M.32	Plant Protection and Instr. System	32
M.32.01	Investment Protection	32-1
M.32.02	Safety Protection	32-2
M.32.03	Special Nuclear Area Instr.	32-3
M.34	Fuel Handling Storage & Shipping System	34
M.34.13	Core Refueling	34-13
M.34.14	Site Fuel Handling	34-14
M.34.87	Spent Fuel Storage Cooling	34-87
M.37	Plant Control, Data & Instr. System	37
M.37.07	Plant Supv. Control	37-7
M.37.01	NSSS Control	37-1
M.37.06	BOP Control	37-6
M.37.08	Plant Oper. Supt.	37-8
M.37.35	Data Processing	37-35
M.56	Reactor Cavity Cooling System	56
M.57	Shutdown Cooling System	57
M.57.01	Shutdown Circulator	57-1
M.57.02	Shutdown Cooling Heat Exchanger	57-2
M.57.03	Shutdown Cooling Heat Removal Control	57-3
M.57.04	Shutdown Cooling Water	57-4

III. BY NUMERICAL SEQUENCE

The final number provides the numerical sequence of the DDN in the series identified by the preceding designators.

Examples are:

M.02.01 is availability and maintenance data that applies to the overall modular HTGR plant and is the first such DDN.

M.10.18.06 is applicable to System 10, Subsystem 18 for the modular HTGR plant, and is the sixth such DDN.

Each DDN is assigned a priority which is based on the following indices;

- Urgency.
- Cost Benefit.
- Uncertainty in Existing Data.
- Importance of New Data.

The Urgency is a measure of the final schedular date that the data is required less the length of time needed to obtain the data. For example, if data is required in 5 years and it is estimated that it will take approximately 2 years to obtain the data, then the Urgency figure of merit would be 5 - 2 = 3. Using this approach, tasks that should be started in Fy-87 are rated 1. The numerical rating would increase progressively reflecting required start dates in later years.

The Cost-Benefit of performing the work required to satisfy the need is a measure of the impact on a single plant cost relative to the development cost as indicated on a scale of high, medium, and low. A high Cost-Benefit is defined when the savings exceed the DDN development cost by a factor of greater than ten (>10). Medium or low ratings reflect lesser Cost-Benefit ratios. Note that the plant savings reflect both cost and schedule impacts.

Uncertainty in Existing Data expresses the designer's lack of confidence in the available data on which the conceptual design is based, on a scale of high, medium, and low.

Importance of New Data expresses the significance or effect of the new data on the design including consideration of available back-up solutions, on a scale of high, medium, and low.

The schedule for the DDNs is based on completing conceptual design by the end of FY-87. Preliminary and final design phases of 2- and 4-year durations, respectively, are assumed to follow. This schedule is shown in Fig. 1, which also identify other key design and licensing dates.

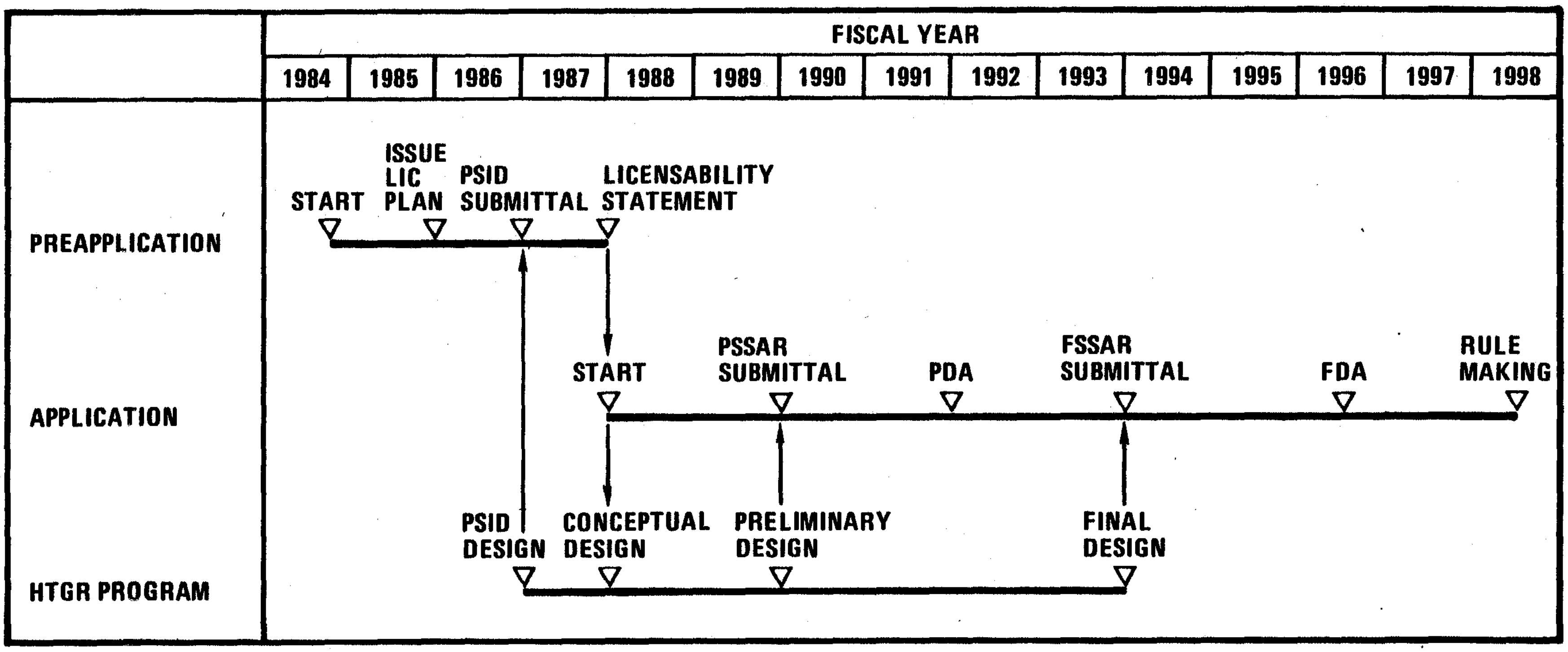

Fig 1. Schedule For MHTGR Design And Licensing Activities.

A listing of the current DDNs, ordered by DDN number, is presented in Table 2. Table 2 also contains pertinent cost information and identifies its sources.

4. REFERENCES

1. "Functional Analysis Report 4 x 350 MW(t) Module HTGR Plant," Issued by GA Technologies, DOE-HTGR-86-02, Rev. 2 (PC-000207/2), February 1987.

2. Gorholt, W., "Graphite Technology Development Plan," GA Technologies, DOE-HTGR-86-037, Rev. 1 (PC-000213/2), March 1987.

3. Hanson, D., "350 MW(t) Modular HTGR-Fuel/Fission Product Technology Developent Program Plan," GA Technologies, HTGR-86-027, Rev. 0, (PC-000215/0), September 1986.

4. Betts, W. S., "Metals Technology Development Plan - 4 x 350 MW(t) Modular HTGR," GA Technologies, DOE-HTGR-86-087, Rev. 2, (PC-000210/4), March 1987.

5. "Regulatory Technology Development Plan for the Standard Modular High-Temperature Gas-Cooled Reactor", Issued by GA Technologies DOE-HTGR-86-064, January 1987.

TABLE 2
LISTING OF DESIGN DATA NEEDS (DDNs)

PREPARED BY MHTGR PROJ
PLANNING & SCHEDULING

* ESTIMATE SOURCE LEGEND:
1-MOD. HTGR TEST PROJ DEF STUDY, 2/25/86
2-MHTGR (GA INTERNAL EST), 3/21/86
3-GRAPHITE TECH PLAN, 3/21/86
4-FUEL/FISSION PRODUCT TECH DEV PROG PLAN, 9/30/86
5-METALS TECH DEV PLAN (PRELIM LTR J. JONES TO AJN, 0630-56-86, 6/30/86)
6-TELECON GA (VF) TO BNI (TS), 5/30/86 & 7/24/86)
7-HTGR (GA INTERNAL EST) 7/24/86
8-METALS TECH DEV PLAN (PRELIM ORNL TELECON DATA 7/24/86)
9-ORNL LETTER 1110-108-86, 11/10/86
10-GA INTERNAL ESTIMATE, 3/27/87
11-ORNL LETTER L0571, 1/16/87
12-ORNL TELEX (M. KANIA TO D. HANSON) 3/19/87
**-DDN M.10.17.28 & M.21.02.03 ESTIMATES ARE INTERDEPENDENT

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
	M.05 SAFETY AND RELIABILITY				
M.05.01	VALIDATION OF DIFFERENTIAL PRESSURE AND SHEAR FORCE RATIO MODELS DURING DEPRESSURIZATION ACCIDENT	9	ORNL	400	0
	SUBTOTAL			400	0
	M.07 FUEL/FISSION PRODUCT				
M.07.01	FISSION GAS RELEASE FROM CORE MATERIALS	4	ORNL	960	0
M.07.02	FISSION METAL EFFECTIVE DIFFUSIVITIES IN FUEL KERNELS	4	ORNL	350	0
M.07.03	FISSION PRODUCT EFFECTIVE DIFFUSIVITIES IN PARTICLE COATINGS	4	ORNL	400	0
M.07.04	FISSION PRODUCT DIFFUSIVITIES/SORPTIVITIES IN GRAPHITE	4	ORNL	2880	30
M.07.05	H3 TRANSPORT IN CORE MATERIALS	4	ORNL	550	0
M.07.06	FISSION PRODUCT DEPOSITION CHARACTERISTICS FOR STRUCTURAL METALS	4	ORNL	3325	255

TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
M.07.07	FISSION PRODUCT REENTRAINMENT CHARACTERISTICS FOR STRUCTURAL METALS	4	ORNL	IN.07.06	0
M.07.08	FISSION PRODUCT WASHOFF CHARACTERISTICS FOR STRUCTURAL METALS	4	ORNL	550	55
M.07.09	CHARACTERIZATION OF THE EFFECT OF DUST ON FISSION PRODUCT TRANSPORT	4	ORNL	970	180
M.07.10	TRITIUM PERMEATION OF STEAM GENERATOR TUBES	4	ORNL	80	0
M.07.11	FISSION PRODUCT TRANSPORT IN REACTOR BUILDING DURING CORE CONDUCTION COOLDOWN TRANSIENTS	12	ORNL	350	0
M.07.12	VALIDATION OF DESIGN METHODS FOR FISSION GAS RELEASE	4	ORNL	2470	0
M.07.13	VALIDATION OF DESIGN METHODS FOR FISSION METAL RELEASE	4	ORNL	IN.07.12	0
M.07.14	VALIDATION OF DESIGN METHODS FOR PLATEOUT DISTRIBUTION	11	ORNL	5670	0
M.07.15	VALIDATION OF DESIGN METHODS FOR FISSION PRODUCT LIFTOFF	11	ORNL	IN.07.14	0
M.07.16	VALIDATION OF DESIGN METHODS FOR FISSION PRODUCT WASHOFF	11	ORNL	IN.07.14	0
M.07.17	UF6-UO3 CONVERSION PROCESS DEVELOPMENT	4	GA	555	780
M.07.18	UCO FISSILE KERNEL PROCESS DEVELOPMENT	4	GA	800	150
M.07.19	FUEL PARTICLE COATING PROCESS DEVELOPMENT	4	GA	625	0
M.07.20	DEVELOPMENT OF PROCESSES FOR FUEL ROD COMPACT FABRICATION	4	GA	1400	300
M.07.21	QC TEST TECHNIQUES DEVELOPMENT	4	GA	330	150
M.07.22	UCO AND THO2 FUEL SCRAP AND WASTE HANDLING DEVELOPMENT	4	GA	1550	450
M.07.23	THO2 FERTILE KERNEL PROCESS DEVELOPMENT	4	GA	835	700
M.07.24	FUEL PROOF TEST	4	ORNL	2230	100
M.07.25	DEVELOPMENT OF PERFORMANCE MODELS FOR DEFECTIVE PARTICLES	4	ORNL	460	0
M.07.26	VALIDATION OF FUEL PERFORMANCE MODELS UNDER NORMAL OPERATING CONDITIONS	4	ORNL	3140	280
M.07.27	VALIDATION OF FUEL PERFORMANCE MODELS UNDER CORE CONDUCTION COOLDOWN CONDITIONS	4	ORNL	3870	0
M.07.28	CHARACTERIZE FUEL COMPACT DIMENSIONAL CHANGE PROPERTIES	10	GA	175	0
	SUBTOTAL			34525	3430
	M.08 DECAY HEAT REMOVAL				
M.08.01	DETERMINE CONDUCTION COOLDOWN TO RCCS	1	GA/BNI	5000	0
	SUBTOTAL			5000	0

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TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
	M.10 REACTOR SYSTEM				
M.10.01	VALIDATION OF DESIGN METHODS FOR GRAPHITE CORROSION	3	ORNL	150	0
	SUBTOTAL			150	0
	M.10.12 NEUTRON CONTROL				
M.10.12.01	CONTROL ROD DRIVE(CRD) DESIGN VERIFICATION	1	GA	280	0
M.10.12.02	RESERVE SHUTDOWN EQUIPMENT (RSE) DESIGN VERIFICATION	1	GA	280	0
M.10.12.03	PRELIMINARY QUALIFICATION OF ELECTROMECHANICAL COMPONENTS OF THE NEUTRON CONTROL ASSEMBLY	1	GA	110	0
M.10.12.04	DEMONSTRATION OF REMOTE HANDLING AND MAINTENANCE FEATURES OF NEUTRON CONTROL ASSEMBLY	1	GA	280	0
M.10.12.05	NEUTRON CONTROL ASSEMBLY FLOW AND LEAK DESIGN VERIFICATION	1	GA	140	0
M.10.12.07	ELECTROMECHANICAL COMPONENTS QUALIFICATION FOR THE NEUTRON CONTROL ASSEMBLY	1	GA	100	0
M.10.12.08	NEUTRON CONTROL ASSEMBLY SEISMIC QUALIFICATION	1	GA	230	0
M.10.12.09	GUIDE TUBES FLOW INDUCED VIBRATION DESIGN VERIFICATION	1	GA	300	0
M.10.12.10	DETERMINE TYPE OF IN-CORE FLUX MONITOR UNIT	1	GA	320	30
M.10.12.11	VERIFY CONTROL ROD INSTRUMENTATION & CONTROL	10	GA	170	50
M.10.12.12	VERIFY STARTUP NEUTRON DETECTOR & CABLING	10	GA	300	50
	SUBTOTAL			2510	130
	M.10.17 REACTOR INTERNALS				
M.10.17.01	UNIAXIAL STRENGTH DATA BASE FOR CORE SUPPORT GRAPHITE	3	ORNL	600	0
M.10.17.02	UNIAXIAL STRENGTH DATA BASE FOR PERMANENT REFLECTOR GRAPHITE	3	ORNL	IN .17.01	
M.10.17.03	MULTIAXIAL STRENGTH OF GRAPHITE FOR CORE SUPPORT	3	ORNL	200	0
M.10.17.04	MULTIAXIAL STRENGTH OF GRAPHITE FOR PERMANENT REFLECTOR	3	ORNL	IN .17.03	
M.10.17.05	FATIGUE STRENGTH OF GRAPHITE FOR CORE SUPPORT COMPONENTS	3	ORNL	600	0
M.10.17.06	FATIGUE STRENGTH OF GRAPHITE FOR PERMANENT REFLECTORS	3	ORNL	IN .17.05	
M.10.17.07	MINER'S LAW FOR GRAPHITE FOR CORE SUPPORT COMPONENTS	3	ORNL	IN .17.05	
M.10.17.08	MINER'S LAW FOR GRAPHITE FOR PERMANENT SIDE REFLECTORS	3	ORNL	IN .17.05	
M.10.17.09	ELASTIC PROPERTIES DATA BASE FOR CORE SUPPORT GRAPHITE	3	ORNL	IN .17.01	
M.10.17.10	ELASTIC PROPERTIES DATA BASE FOR PERMANENT REFLECTOR GRAPHITE	3	ORNL	IN .17.01	
M.10.17.11	IRRADIATION EFFECTS ON MECHANICAL PROPERTIES OF CORE SUPPORT GRAPHITE	3	ORNL	600	0

TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
M.10.17.12	IRRADIATION EFFECTS ON MECHANICAL PROPERTIES OF PERMANENT REFLECTOR GRAPHITE	3	ORNL	IN .17.11	
M.10.17.13	THERMAL PROPERTIES OF CORE SUPPORT GRAPHITE	3	ORNL	185	0
M.10.17.14	THERMAL PROPERTIES OF PERMANENT REFLECTOR GRAPHITE	3	ORNL	IN .17.13	
M.10.17.15	IRRADIATION EFFECTS ON THERMAL PROPERTIES OF CORE SUPPORT GRAPHITE	3	ORNL	IN .17.11	
M.10.17.16	IRRADIATION EFFECTS ON THERMAL PROPERTIES OF PERMANENT SIDE REFLECTOR GRAPHITE	3	ORNL	IN .17.11	
M.10.17.17	CORROSION CHARACTERISTICS OF CORE SUPPORT GRAPHITE	3	ORNL	500	0
M.10.17.18	CORROSION CHARACTERISTICS OF PERMANENT REFLECTOR GRAPHITE	3	ORNL	IN .17.17	
M.10.17.21	CONFIRM LARGE SIZE GRAPHITE FOR PERMANENT REFLECTOR	3	ORNL	20	0
M.10.17.22	NDE DATA FOR REACTOR INTERNALS GRAPHITE SPECIFICATIONS	3	ORNL	50	0
M.10.17.23	CONFIRM STRENGTH OF GRAPHITE CORE SUPPORT	1	GA	250	0
M.10.17.25	CONFIRM HOT DUCT INTEGRITY	1	GA	2250	250
M.10.17.26	DETERMINE EFFECTS OF IRRADIATION ON PROPERTIES OF ALLOY 800H	5	ORNL	2045	260
M.10.17.28	DETERMINE EFFECTS OF PRIMARY COOLANT CHEMISTRY AND TEMPERATURE ON ALLOY 800H	5	ORNL	**2420	835
M.10.17.29	FIBROUS INSULATION MATERIAL PROPERTIES	1	GA	450	50
M.10.17.30	VALIDATE PRESSURE DROP FROM COLD DUCT ENTRANCE TO CORE INLET	10	GA	500	0
	SUBTOTAL			10670	1395
	M.10.18 REACTOR CORE				
M.10.18.01	MULTIAXIAL STRENGTH OF GRAPHITE FOR CORE COMPONENTS	3	ORNL	700	0
M.10.18.02	FATIGUE DATA FOR GRAPHITE FOR CORE COMPONENTS	3	ORNL	175	0
M.10.18.03	STATISTICS OF MECHANICAL PROPERTIES OF GRAPHITE CORE COMPONENTS	3	ORNL	550	0
M.10.18.04	STATISTICS OF IRRADIATION-INDUCED STRAIN OF GRAPHITE FOR CORE COMPONENTS	3	ORNL	250	0
M.10.18.05	STATISTICS OF IRRADIATION-INDUCED CREEP OF GRAPHITE CORE COMPONENTS	3	ORNL	4000	0
M.10.18.06	STATISTICS OF THERMAL PROPERTIES OF GRAPHITE FOR CORE COMPONENTS	3	ORNL	240	0
M.10.18.07	STATISTICS OF FRACTURE MECHANICS PROPERTIES OF GRAPHITE FOR CORE COMPONENTS	3	ORNL	800	0
M.10.18.08	CORROSION CHARACTERISTICS OF CORE COMPONENTS GRAPHITE	3	ORNL	950	0
M.10.18.09	CORROSION EFFECTS ON CORE COMPONENT GRAPHITE DESIGN PROPERTIES	3	ORNL	IN .18.08	
M.10.18.10	DESTRUCTIVE/NDE DATA FOR CORE GRAPHITE SPECIFICATIONS	3	ORNL	50	0

TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
M.10.18.11	VALIDATE FUEL ELEMENT DYNAMIC STRENGTH PREDICTIONS	1	GA	1940	500
M.10.18.12	VALIDATE FUEL ELEMENT FAILURE MODE PREDICTIONS	1	GA	1110	100
M.10.18.13	VALIDATE CORE COMPONENT SEISMIC LOAD PREDICTIONS	1	GA	1400	150
M.10.18.14	VALIDATE CONTROL ROD SHOCK ABSORBER CHARACTERISTICS	1	GA	450	0
M.10.18.15	VALIDATE CONTROL CHANNEL FLOW PREDICTIONS	1	GA	500	0
M.10.18.16	VALIDATE FUEL ELEMENT CHANNEL FLOW PREDICTIONS	1	GA	250	0
M.10.18.17	VALIDATE CONTROL ROD VIBRATION PREDICTIONS	1	GA	350	0
M.10.18.18	VALIDATE CORE CROSSFLOW PREDICTIONS	1	GA	500	0
M.10.18.19	CONFIRM ABSENCE OF CORE FLUCTUATIONS	1	GA	700	0
M.10.18.20	CONFIRM ABSENCE OF FUEL/REFLECTOR COLUMN FLOW INDUCED VIBRATION	7	GA	500	0
M.10.18.21	RESERVE SHUTDOWN & BURNABLE POISON MATERIAL PROCESS DEVELOPMENT	10	GA	190	0
M.10.18.22	CORROSION CHARACTERISTICS OF COATED B4C	10	GA	400	50
M.10.18.23	CORROSION CHARACTERISTICS OF CORE MATRIX MATERIALS	10	GA	500	50
M.10.18.24	VALIDATE THE PRESS DROP & FLOW MIXING IN THE LOWER REFLECTOR /CORE SUPPORT BLOCKS	10	GA	350	0
M.10.18.25	VALIDATE THE PRESS DROP & FLOW DISTRIBUTION THROUGH THE METALLIC PLENUM ELEMENT & TOP REFLECTOR	10	GA	350	0
	SUBTOTAL			17205	850
	M.11 VESSEL SYSTEM				
	M.11.06 VESSEL AND DUCTS				
M.11.06.01	NIL-DUCTILITY TRANSITION TEMPERATURE SHIFT FOR REACTOR VESSEL MATERIAL IRRADIATED AT LOW TEMPERATURES (C-E)	5	ORNL	350	0
M.11.06.02	DETERMINE PROPERTIES OF SA 533B (Mn-1/2 Mo-1/2 Ni) BASE METAL & WELDMENT AT ELEVATED TEMPERATURES (C-E)	8	ORNL	150	0
	SUBTOTAL			500	0

TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
	M.20 REACTOR SERVICES GROUP				
	M.20.16 REACTOR SERVICE EQUIP. & STOR. WELLS				
M.20.16.01	REACTOR EQUIPMENT SERVICE FACILITY TOOLS DESIGN VERIFICATION	1	GA	260	0
M.20.16.02	RESERVE SHUTDOWN VACUUM TOOL DESIGN VERIFICATION	1	GA	120	0
	SUBTOTAL			380	0
	M.21 HEAT TRANSPORT SYSTEM				
M.21.00.01	DETERMINE CORE EXIT PLENUM AND HOT DUCT FLOW FIELD	1	GA	1790	0
	SUBTOTAL			1790	0
	M.21.01 MAIN CIRCULATOR				
M.21.01.01	MAIN CIRCULATOR MAGNETIC AND CATCHER BEARINGS DESIGN VERIFICATION	1	GA	1490	0
M.21.01.02	MAIN CIRCULATOR PROTOTYPE DESIGN VERIFICATION	1	GA	6500	1500
M.21.01.03	MAIN CIRCULATOR MOTOR COOLING DESIGN VERIFICATION	1	GA	300	0
	SUBTOTAL			8290	1500
	M.21.02 STEAM GENERATOR				
M.21.02.02	DETERMINE PROPERTIES OF 2-1/4CR-1MO BASE METAL AND WELDMENT(CE)	5	ORNL	2485	280
M.21.02.03	DETERMINE PROPERTIES OF ALLOY 800H BASE METAL AND WELDMENTS(CE)	5	ORNL	**1280	10
M.21.02.04	TUBE BUNDLE ACCOUSTIC TEST (CE)	1	C-E	200	0
M.21.02.07	LARGE HELICAL COIL PROGRAM (CE)	1	C-E	270	0
M.21.02.08	AIR FLOW TEST (CE)	1	C-E	380	0
M.21.02.10	STEAM GENERATOR INSULATION VERIFCIATION TESTS (CE)	1	C-E	155	0
M.21.02.11	VIBRATION FRETTING WEAR & SLIDING WEAR PROTECTION TESTS (CE)	1	C-E	1600	0
M.21.02.12	TUBE WEAR PROTECTION DEVICE TESTS (CE)	1	C-E	420	0
	SUBTOTAL			6790	290

TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
	M.30 MISC CONTROL & INSTR. GROUP				
	M.30.01 NSSS ANALYTICAL INSTRUMENTATION SYSTEM				
M.30.01.01	VERIFY NSSS ANALYTICAL INSTRUMENTATION SYSTEM	10	GA	500	350
	SUBTOTAL			500	350
	M.32 PLANT PROTECTION & INSTRUMENTATION SYS				
	M.32.02 SAFETY PROTECTION				
M.32.02.01	VERIFY PLANT PROTECTION AND INSTRUMENTATION SYSTEM MOISTURE MONITOR	1	GA	660	160
M.32.02.02	VERIFY SHUTDOWN COOLING HEAT EXCHANGER LEAK DETECTION	10	GA	350	90
M.32.02.03	VERIFY HELIUM MASS FLOW INSTRUMENTATION	1	GA	240	90
M.32.02.04	VERIFY STEAM GENERATOR INLET HELIUM TEMPERATURE INSTRUMENTATION	1	GA	310	110
M.32.02.05	VERIFY PLANT PROTECTION AND INSTRUMENTATION SYSTEM (PPIS) SURVEILLANCE TESTING	1	GA	300	70
	SUBTOTAL			1860	520
	M.34 FUEL HANDLING STORAGE & SHIPPING SYSTEM				
	M.34.13 CORE REFUELING				
M.34.13.01	FUEL HANDLING MACHINE (FHM) HANDLING MECHANISM AND GRAPPLE DESIGN VERIFICATION	1	GA	1195	70
M.34.13.02	FUEL TRANSFER CASK COMPONENT DESIGN VERIFICATION	1	GA	320	70
M.34.13.03	PLUG ACTUATOR AND TURNTABLE ASSEMBLY COMPONENT DESIGN VERIFICATION	1	GA	200	70
M.34.13.05	VERIFY FUEL HANDLING SYSTEM INSTRUMENTATION & CONTROLS	1	GA	300	70
	SUBTOTAL			2010	280

TABLE 2 (CONT.)

	DESCRIPTION	* ESTIMATE SOURCE	RESP COST EST ORG	CURRENT OPERATING ESTIMATE ($000)	CAPITAL ($000)
	M.37 PLANT CONTROL DATA & INSTR. SYSTEM				
	M.37.01 NSSS CONTROL				
M.37.01.01	VERIFY NSSS CONTROL SYSTEM WITH SIMULATOR	1	GA	760	0
M.37.01.02	VERIFY NSSS CONTROL LAYOUT	1	GA	180	30
M.37.01.03	VERIFY OF CRT DISPLAYS OF NSSS MODULE OPERATING DATA	1	GA	160	0
	SUBTOTAL			1100	30
	M.57 SHUTDOWN COOLING SYSTEM				
	M.57.01 SHUTDOWN CIRCULATOR				
M.57.01.01	SHUTDOWN CIRCULATOR MOTOR COOLING DESIGN VERIFICATION	1	GA	300	0
M.57.01.02	SHUTDOWN CIRCULATOR PROTOTYPE DESIGN VERIFICATION	1	GA	2990	800
	SUBTOTAL			3290	800
	TOTAL			96970	9575

15

[72-1]

DATE: 3/4/87

VALIDATION OF DIFFERENTIAL PRESSURE AND SHEAR FORCE
RATIO MODELS DURING DEPRESSURIZATION ACCIDENT
DDN M.05.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 05

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

. Design methods and codes must be validated to accurately calculate transient differential pressures across components during depressurization accidents. The design methods and codes used to predict the extent to which fission products are lifted off from the primary circuit must be validated.

1.1 Summary of Function/Title/Assumptions

F3.0 "Maintain Control of Radionuclide Release," Assumption 4: Data and models are available to adequately assess safety risk envelope such that mean risk satisfies safety risk goal.

F3.1.1.2.2 "Control Transport from Primary Circuit," Assumption 1C: Validated methods will be available to describe the reentrainment and redeposition of plateout activity in the primary circuit to within a factor of [10x] at 95% confidence.

1.2 Current Data Base Summary

The transient thermal and fluid flow behavior of the primary coolant during a depressurization accident is calculated using the RATSAM computer code. Differential pressures across components within the primary coolant boundary, as calculated by RATSAM, have been validated against one experimental rapid depressurization test of the Calder Hall Stage 1 gas reactor model. RATSAM calculations of temperature, pressure, and flow have been compared to data taken from CPL 2/4 experiments.

Lift off involves the mechanical removal of fission products from the plated out surfaces of reactor components. Lift off is associated with high velocity helium flow. The liftoff of condensible fission products is currently modeled as a function of shear force ratio during a depressurization. That is, liftoff is correlated with the ratio of local shear force during the depressurization to local shear force during normal operation. The local shear force ratios are determined based on the ratios of flow rates, per fundamental theoretical relationships for one-dimensional tubes. The transient flow rates are modeled in the RATSAM code.

1.3 Data Needed

Data are needed to validate the RATSAM model of transient shear force ratios during a depressurization, over a range of representative geometries. Data are needed to characterize the shear forces that act on walls under normal operation conditions and under conditions of accelerated flow that occur due to primary system depressurization. Various wall/flow configurations should be studied, including those representative of flow past steam generator tubes, flow through a vessel and shroud annulus, flow in a plenum where the gas turns 90° from entrance to exit, flow through a tube, and flow past a relatively irregular surface such as a circulator.

Data are needed to further validate the differential pressure calculations performed by the RATSAM code. Data are needed to characterize the differential pressures that occur in a gas flow system due to a depressurization event. The data should cover a range of depressurization rates. Data should be gathered across components with initial flow resistance characteristics representative of MHTGR components, including the core, core support, steam generator, S/G support, and concentric cross ducts. Pressure information should be acquired at multiple locations in a plenum, for information on the importance of local versus average pressure variations. Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The following range of service conditions should be studied:

Reynolds No., initial	10^4 to 5×10^6
Pressure, initial	5.07 to 6.38 MPa (735 to 925 psia)
Gas temperatures	250 to 700°C (482 to 1292°F)
Shear force ratio	0.5 to 3
Depressurization time constant	20 to 900 s

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Use theoretical or empirical correlations as available for various geometries, to determine relationship between expected shear force ratio for that geometry versus calculated shear force ratio (based on tube flow). For any geometries that are treated nonconservatively, apply an appropriate factor for margin when liftoff is calculated. For any geometries that are treated conservatively, acknowledge conservatism in the release calculation. Justify the use of shear force ratios calculated in RATSAM to represent expected shear force ratios, within the uncertainties caused by different geometries, on the basis that the correlations are for the most part empirical.

Validate differential pressure models on the basis of good agreement with the Calder Hall and CPL 2/4 experiments.

2.2 Obtain results of pressure transient calculated by independent computer code. Compare differential pressures with those calculated by RATSAM. Good agreement would validate RATSAM.

3. <u>SELECTED DESIGN APPROACH AND EXPLANATION</u>

Obtain experimental data on shear force ratios on various surface geometries for simulated depressurization events. These data will be used to validate shear force ratio calculations performed by the RATSAM code. These calculations are otherwise verified but not experimentally validated if Alternative 2.1 is selected.

Obtain experimental data on pressure transients during range of depressurization events, to validate differential pressure calculations performed by RATSAM code. Validation by experimental data is a more direct method than by comparison to results from another code (Alternative 2.2). Validation against multiple experiments that cover a range of depressurization conditions is more convincing than validation against two experiments that do not adequately cover the range (Alternative 2.1).

4. <u>SCHEDULE REQUIREMENTS</u>

Validation of differential pressure and shear ratio models is required prior to the start of the final design (9/89).

5. <u>PRIORITY</u>

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. <u>FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION</u>

The fallback position for shear force ratio calculations is Alternative 2.1: To make the case that shear force ratios calculated based on one-dimensional tube geometry are sufficiently representative of expected shear force ratios, and to account for differences via an uncertainty factor on the resultant liftoff. The consequence of using such conservatisms may be that plateout criteria will be unnecessarily restrictive, resulting in increased plant cost due to the requirement for tighter fuel quality specifications or limitation on the leak area size by addition of flow restrictors.

The fallback position for differential pressure calculations is Alternative 2.2: To compare RATSAM results with pressure transient results calculated by an independent computer code. Good agreement with an independent code, coupled with good agreement with the two experiments (Alternative 2.1) may not satisfy licensing authorities.

Gayle Cadwallader 3/13/87
Originator Date

031387
Department Manager Date

for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION GAS RELEASE FROM CORE MATERIALS
DDN M.07.01
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The dominant sources of fission gas release are heavy-metal contamination in the fuel-compact matrix and failed fuel particles with exposed kernels; consequently, the release characteristics of these two sources must be determined, including the effects of environmental and irradiation conditions, for normal operating conditions, for wet shutdown conditions, and for core conduction cooldown transients.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 3: Failed fuel particles will hydrolyze during irradiation.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 6: Reference correlations for fission gas release from heavy metal contamination are accurate to within [4x] at 95% confidence.

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 3: Exposed fuel kernels will hydrolyze during irradiation.

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 4: Reference correlations for transport of fission products in fuel kernels are accurate to within [TBDx] at 95% confidence.

F2.1.4.1.1.2.1.1.1 "Protect the Capability to Retain Radionuclides in Fuel Kernels," Assumption 1: Reference correlations describe fission product release from kernels under core conduction cooldown conditions to within a factor of [TBD] at 95% confidence.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 4: No incremental release from heavy metal contamination as a result of steam ingress.

F3.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 1: Validated methods are available to adequately determine release of fission gases under wet shutdown conditions.

F3.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 3: Reference correlations describe fission product release from kernels under core conduction cooldown conditions to within a factor of [TBD] at 95% confidence.

F3.1.1.2.1.2 "Retain Radionuclides in Core Graphite," Assumption 2B: Iodines may be partially retained by core graphite under wet shutdown conditions but insufficient data quantify effect.

1.2 Current Data Base Summary

The present data base for fission gas release from heavy-metal contamination and from failed particles is derived primarily from TRIGA measurements on fuel compact matrix doped with uranium and on laser-failed fuel particles, respectively. The effects of fuel hydrolysis (reaction of exposed kernels with water) on gas release are derived from laboratory measurements and short-term TRIGA tests. Isothermal, in-pile hydrolysis tests on reference fuel (HRB 17/18) were recently completed at ORNL, and the temperature dependence of gas release from both unhydrolyzed and hydrolyzed fuel will be addressed in the planned HFR B1 test.

The extent to which the fuel must retain fission gases under dry and wet conditions is specified in the functions cited above. The present data base is inadequate to ascertain whether or not the selected design meets these requirements at the specified confidence level, especially with regard to the extent and consequences of hydrolysis under irradiation and during wet core conduction cooldown transients.

The present data base for fission gas release from failed particles under core conduction conditions is derived largely from measurements on laser-failed HEU UC_2/ThO_2 particles; the iodine release data are exclusively from this source.

1.3 Data Needed

Measurement of the fission gas release rates (Kr, Xe, I, and Te) from heavy-metal contamination and from failed reference fuel particles as a function of temperature, half-life, burnup and flux under irradiation, under wet shutdown conditions, and under dry and wet core conduction cooldown conditions. In addition, the effect of hydrolysis on gas release must be quantified for steady-state irradiation and for transient wet shutdown and wet core conduction cooldown conditions. The assumption that I isotopes behave like Xe isotopes also must be confirmed. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Fuel Operating Temperature	700 - 1250°C

Maximum Fissile Particle Burnup	22% FIMA
Maximum Fertile Particle Burnup	3% FIMA
Maximum Fast Fluence (E > 29 fJ)	5×10^{25} n/m^2
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	1 atm
<u>Wet Shutdown Conditions</u>	
Environment	He/H_2O/Air
Fuel Temperature Range	[100 - 300] °C
Coolant Pressure	1 atm
Range of Coolant Impurity Levels	[0.01 - 1] atm H_2O [TBD] atm O_2 [TBD] atm N_2
Fission Products of Interest	I > Kr, Xe
<u>Core Conduction Cooldown Transients</u>	
Environment	He; He/CO/H_2; CO/N_2
Fuel Temperature Range Pressurized Cooldown Depressurized Cooldown	 900 - 1200°C 1200 - 1800°C
Pressure	1 atm
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
Fission Products of Interest	I > Xe > Kr

2. DESIGNER'S ALTERNATIVES

The following alternative has been considered:

1. Use extrapolated HEU UC_2 data base and models, recognizing that the data on the effects of hydrolysis are based primarily on laboratory

and TRIGA measurements which may significantly overestimate the in-pile effects.

2. Use FRG UO_2 data base and models, recognizing that the effects of hydrolysis on UO_2 are probably less than for UCO.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure fission gas release (Kr, Xe, and I) from LEU UCO/ThO_2 TRISO fuel irradiated under near normal HTGR flux over a range of temperatures and under dry and wet conditions. Measure the fission gas from irradiated reference fuel under pressurized and depressurized core conduction cool-down conditions. Such measurements will reduce the uncertainties in the fission gas retention characteristics of the reference fuel and provide a basis for judging the adequacy of the present design.

4. SCHEDULE REQUIREMENTS

Preliminary data by 3/89 (6 months prior to PSSAR); final results by 9/92 (1 year prior to FSSAR).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Use alternative 1 with more restrictive limits on as-manufactured fuel quality and in-service failure to compensate for conservatism in the fission gas release models. The consequences would be higher fuel development costs and higher fuel manufacturing costs associated with unnecessarily tight as-manufactured fuel quality requirements.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION METAL EFFECTIVE DIFFUSIVITIES IN FUEL KERNELS
DDN M.07.02
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel kernel of the coated particle is the initial barrier to the release of fission metals from the core and may provide significant holdup; consequently, the transport properties of fission metals in the reference UCO/ThO_2 kernels must be characterized for normal operating conditions and for core conduction cooldown transients.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 4: Reference correlations for fission product transport in fuel kernels are accurate to within a factor of [TBD] at 95% confidence.

F2.1.4.1.1.2.1.1.1 "Protect the Capability to Retain Radionuclides in Fuel Kernels," Assumption 1: Reference correlations for fission product transport in fuel kernels are accurate to within a factor of [TBD] at 95% confidence.

1.2 Current Data Base Summary

The present data base is derived primarily from measurement on particles irradiated in accelerated capsules. There are some FRG data for Cs, Sr, and Ag in oxide kernels of intact particles which were irradiated under near real-time conditions as well as limited laboratory data on Cs release from ThO_2 kernels. Limited data are available for the release of Pu from failed particles.

1.3 Data Needed

Correlations are needed for the effective diffusivities of key fission metals (Cs, Ag, and Sr) in LEU UCO and ThO_2 kernels as a function of temperature, burnup and, if appropriate, neutron flux for normal operation and core conduction cooldown conditions. The tentative observation that the metal diffusivities in the kernels of intact particles are significantly lower than in the kernels of failed particles also needs to be confirmed and quantified. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Fuel Operating Temperature	700 - 1250°C
Maximum Fissile Particle Burnup	22% FIMA
Maximum Fertile Particle Burnup	3% FIMA
Maximum Fast Fluence (E > 29 fJ)	5×10^{25} n/m^2
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm O_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	1 atm
Fission Products of Interest	Cs, Ag >> Sr

Core Conduction Cooldown Transients

Environment	He; He/CO/H_2; CO/N_2
Fuel Temperature Range Pressurized Cooldown Depressurized Cooldown	 900 - 1200°C 1200 - 1800°C
Pressure	1 atm
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
Fission Products of Interest	Sr > Ag, Cs

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use current models which assume that irradiation conditions, particularly high neutron fluxes and high temperatures, have no special effects on kernel release.

2. Use FRG correlations for kernel diffusivities.

3. Take no credit for kernel retention when calculating fission metal release rates from core.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Complete measurement and modeling of fission metal release from reference fuel kernels in failed and intact particles under near real-time irradiation and core conduction cooldown conditions. The estimated uncertainties in the reference correlations are excessively large; one major source of uncertainty is that these correlations are based largely on data from accelerated irradiation tests which may significantly overestimate kernel release under real-time conditions.

4. SCHEDULE REQUIREMENTS

Preliminary data by 3/89 (6 months prior to PSSAR); final data by 9/92 (1 year prior to FSSAR).

5. PRIORITY

Urgency: 1
Cost benefit: L
Uncertainty in existing data: H
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 1 along with a more conservative fuel and core design to account for the uncertainties resulting from deriving the retention characteristics of oxiditic kernels from accelerated irradiation data. Failure to fully exploit the inherent retentivity of oxide-based TRISO particles will necessitate more reliance upon the core graphite as a barrier to release of fission metals.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

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G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION PRODUCT EFFECTIVE DIFFUSIVITIES IN PARTICLE COATINGS
DDN M.07.03
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel particle coatings, particularly the SiC coating, are the primary barrier to release of fission products from the core during normal operation and during core conduction cooldown transients; consequently, the transport properties of fission products in particle coatings must be determined as a function of environmental and irradiation conditions.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 6: Reference correlations are adequate to describe fission product transport in SiC and PyC coatings to within factor of [TBD] at 95% confidence.

F2.1.4.1.1.2.1.1.2 "Protect the Capability to Retain Radionuclides with Particle Coatings," Assumption 4: Reference correlations for radionuclide transport in particle coatings are accurate to within [TBD] at 95% confidence.

F3.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 3: Reference correlations for radionuclide transport in particle coatings are accurate to within [TBD] at 95% confidence.

1.2 Current Data Base Summary

The present data base resulted largely from diffusivity measurements for various fission products in SiC and pyrocarbon coatings in a laboratory environment. These data are supported by limited in-pile data for Cs and Sr inferred from the results of irradiation experiments. There are limited and highly variable data on the diffusive release of fission gases from BISO particles, but the relevance of these data to the transport of gases in the OPyC coatings of TRISO particles is questionable.

1.3 Data Needed

The effective diffusivities of key radionuclides in particle coatings are needed as a function of temperature and, as required, of fluence, irradiation history, and as-manufactured coating attributes for normal operation and for core conduction cooldown conditions; specifically, the effective diffusivities of the volatile fission metals (Ag, Cs, and Sr) in SiC coatings are needed as are the diffusivities of key fission gases (Kr, Xe, and I) in pyrocarbon (PyC) coatings. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Fuel Operating Temperature	700 - 1250°C
Maximum Fissile Particle Burnup	22% FIMA
Maximum Fertile Particle Burnup	3% FIMA
Maximum Fast Fluence (E > 29 fJ)	5×10^{25} n/m^2
Coolant Impurity Levels	126 µatm H_2O 315 µatm CO 126 µatm CO_2 Total Oxidants 630 µatm 630 µatm H_2
Coolant pressure	1 atm
Fission Products of Interest	Ag > Cs > Sr in SiC

Core Conduction Cooldown Transients

Environment	Helium
Fuel Temperature Range	
Pressurized Cooldown	900 - 1200°C
Depressurized Cooldown	1200 - 1800°C
Pressure	1 atm
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
Fission Products of Interest	
PyC	I > Xe > Kr
SiC	Cs > Ag > Sr

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use FRG correlations for FP diffusivities in coatings which were derived from data taken on non-U.S. German particles and assume the data are applicable to U.S. fuel.

2. Assume no retention of Ag and complete retention of Cs and Sr by SiC coatings and assume no gas retention by the OPyC on particles with failed or defective SiC coatings.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Determine the effective diffusivities of Ag, Cs, and Sr in the SiC coatings and of Kr, Xe, and I in the OPyC coatings of irradiated, production-type TRISO particles manufactured to GA fuel product and process specifications. Correlate these diffusivities as a function of temperature and, as appropriate, of fluence, irradiation history, and as-manufactured coating attributes.

The available data suggest that the diffusivities of volatile fission products in SiC and OPyC can be strongly dependent upon the physical structure of the coatings which can be influenced by the coating process parameters and particle irradiation history. Consequently, the most reliable data would be obtained from TRISO particles manufactured to GA product and process specifications and irradiated under conditions representative of modular HTGR cores.

4. SCHEDULE REQUIREMENTS

Final data by 9/92 (one year prior to FSSAR).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 1 along with a more conservative fuel and core design to account for the uncertainties resulting from deriving the retention characteristics of U.S. TRISO particles from German performance data. Failure to fully exploit the inherent retentivity of TRISO particles will necessitate a more conservative fuel and core design which could include unnecessarily restrictive limits on fuel temperatures during core conduction cooldown transients.

[40-11]

DL Hanson 3/19/87

Originator Date

RJ Turner 3/20/87

Department Manager Date

[illegible] for

G. C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 3/27/87

FISSION PRODUCT DIFFUSIVITIES/SORPTIVITIES IN GRAPHITE
DDN M.07.04
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel element graphite can significantly attenuate the release of fission metals from the core during normal operation and during core conduction cooldown transients; consequently, the transport properties of fission metals in graphite must be determined as a function of environmental and irradiation conditions.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1.2 "Retain Radionuclides in Core Graphite," Assumption 3: Transport of radionuclides in core graphite is adequately described by reference correlations to within a factor of 10 at 95% confidence.

F1.1.4.1.1.2.2 "Control Transport in Primary Circuit," Assumption 4: The available data, design methods and computer codes for predicting transport in primary circuit are accurate to within 10x at 95% confidence.

F2.1.4.1.1.2.1.2 "Protect the Capability to Retain Radionuclides in Core Graphite," Assumption 2: Transport of fission metals in core graphite during core conduction cooldown transients is adequately described by reference correlations to within a factor of [TBD] at 95% confidence.

F2.1.4.1.1.2.2 "Protect the Capability to Control Transport in Primary Coolant Circuit," Assumption 5: The design methods and codes for predicting FP transport under core conduction cooldown conditions are accurate to within 10x at 95% confidence.

F3.1.1.2.1.2 "Retain Radionuclides in Core Graphite," Assumption 1b: Transport of fission metals in core graphite [during core conduction cooldown accidents] is adequately described by reference models to within a factor of 10 at 95% confidence.

1.2 Current Data Base Summary

The present correlations for fission metal diffusivities in core graphite are derived largely from laboratory measurements on unirradiated graphites and from profile measurements in various irradiated graphites. The correlations for Cs, Sr, and Pu sorptivities on graphite are derived largely from measurements on unirradiated graphites, but there are limited data for Cs and Sr on irradiated graphite and irradiated fuel compact matrix material. The available data indicate that the transport of Cs, Sr, and Ag in graphite is strongly affected by neutron irradiation. There are limited laboratory data that indicate the vapor pressure of Cs over graphite increases in the presence of coolant impurities and as a consequence of partial graphite oxidation. Ag transport through grapahite may be reduced by elevated pressures.

1.3 Data Needed

Correlations for the diffusivities and sorptivities of Cs, Ag, and Sr in fuel-compact matrix and core graphites as a function of temperature, fluence, and, as appropriate, coolant impurities, system pressure, and the extent of graphite oxidation under normal operating and core conduction cooldown conditions. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Graphite Temperature Range	300 - 1100°C
Maximum Fast Fluence (E > 29 fJ)	5×10^{25} n/m^2
Primary Coolant Temperature Range	300 to 700°C
Coolant Impurity Levels	126 µatm H_2O 315 µatm CO 126 µatm CO_2 Total Oxidants <630 µatm 630 µatm
Coolant Pressure	1 atm
Range of Graphite Burnoff	0.1 - 10%
Fission Products of Interest	Cs, Ag > Sr

Core Conduction Cooldown Conditions

Environment	He; He/CO/H_2; CO/N_2

Graphite Temperature Range	
Pressurized Cooldown	700 - 1200°C
Depressurized Cooldown	1200 - 1800°C
Coolant Pressure Range	[1] atm*
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
Range of Graphite Burnoff	[<0.1 - 10]%
Fission Products of Interest	Sr, Cs > Ag

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use the current reference correlations which have very large uncertainties.

2. Use the reference German correlations for transport in pebble matrix and FRG graphites which do not explicitly treat irradiation or environmental effects.

3. Do not take credit for the core graphite as a barrier to fission metal release and rely exclusively on the SiC coating.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Complete the measurement and modeling of fission metal transport in fuel rod matrix and core graphites, and establish a correlation that explicitly accounts for the effects of temperature, fluence, coolant impurities, graphite burnoff and, if appropriate, system pressure and fission metal concentration. The core graphite should be a very significant barrier to the release of fission metals; however, the reference correlations have very large uncertainties because many of the apparent variables cited above are not treated explicitly and because the correlations are based largely on measurements made on unirradiated graphites.

4. SCHEDULE REQUIREMENTS

Preliminary data by 3/89 (6 months prior to PSSAR); final data by 9/92 (1 year prior to FSSAR).

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*Ag data at P > [10] atm needed on lower priority basis.

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H .

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 1 and accept the very large uncertainties in the reference correlations which, at least in part, result from not explicitly considering irradiation or environmental effects. The risk is that the licensing authorities may not give credit for the substantial attenuation of fission metal release by the core graphite during normal operation and core conduction cooldown accidents. If no credit is taken for the attenuation of fission metal release by the core graphite, the retention requirements imposed upon the fuel particle coatings become correspondingly more stringent.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

~~G. C. Bramblett~~ for 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

H^3 TRANSPORT IN CORE MATERIALS
DDN M.07.05
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Significant quantities of H^3 are produced in the core as a result of ternary fission, neutron activation of Li impurities, and burnout of control materials. However, the H^3 produced from these sources is expected to be largely retained in the core materials (>99%). Moreover, the core graphite is expected to be a major sink for the H^3 produced in the primary coolant by neutron activation of the He^3. Consequently, the transport properties of H^3 in the core materials, especially the H^3 sorptivity of core graphites, must be quantified as a function of irradiation and environmental conditions.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 6: Reference correlations are adequate to describe fission product transport in SiC and PyC coatings to within factor of [TBD] at 95% confidence.

F1.1.4.1.1.2.2 "Control Transport in Primary Circuit," Assumption 6: Core graphite will be major H^3 sink during normal operation.

1.2 Current Data Base Summary

Data on the rates of H^3 release from reference core materials are quite sparse. There are no measurements of the release of H^3 from failed and intact reference UCO/ThO_2 fuel particles. Measurements on various nonreference particle types imply that the kernel release is rapid. An empirical correlation, derived by HRB from annealing data for mixed-oxide TRISO particles, is used to calculate the diffusive release of H^3 from intact TRISO particles. Limited FRG data are available on the transport of H^3 in unirradiated FRG graphites, and the observed sorptivities are relatively small. However, integral H^3 release data from operating HTGRs imply that the effective sorptivity of core graphites may be dramatically increased in the presence of a neutron flux but that H^3 sorbed on core graphites may be desorbed as a consequence of H_2O ingress. Some limited measurements have been made on the retention of H^3 by B_4C pellets, but the effects of irradiation and environment have not been quantified.

1.3 Data Needed

Transport properties of H^3 in SiC coatings and core structural graphites (H451 and Stackpole 2020) are needed as a function of temperature and, if appropriate, fluence, neutron flux and coolant impurity concentrations, especially H_2O and H_2. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Fuel Operating Temperature	700 - 1250°C
Maximum Fissile Particle Burnup	22% FIMA
Maximum Fertile Particle Burnup	3% FIMA
Maximum Fast Fluence (E > 29 fJ)	
H451	5×10^{25} n/m^2
2020	2×10^{24} n/m^2
Graphite Temperature Range	
H451	300 - 1200°C
2020	300 - 750°C
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 <630 μatm 126 μatm H_2
Coolant pressure	1 atm

Wet Shutdown Conditions

Environment	He/H_2O/Air
Fuel Temperature Range	[100 - 300] °C
Coolant Pressure	1 atm
Range of Coolant Impurity Levels	[0.01 - 1.0] atm H_2O [TBD] O_2 [TBD] N_2

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Assume no tritium retention by failed fuel particles, matrix, graphite, and control rod material and limit the amount of

circulating tritium by restricting the allowable Li impurities in core materials and the allowable He^3 impurity in the primary He.

2. Use FRG data for H^3 transport in core materials as available.

3. Assume no tritium retention by failed fuel particles, matrix, graphite, and control rod material and limit the amount of circulating tritium by increasing the capacity of the He purification system.

3. <u>SELECTED DESIGN APPROACH AND EXPLANATION</u>

Measure release of H^3 from failed and intact reference fuel particles as a function of temperature and, as appropriate, of irradiation and environmental conditions. Determine the retentivity of fuel element matrix, core structural graphite, and control materials as a function of temperature and, as appropriate, irradiation and environmental parameters. Unless credit is taken for the H^3 retentivity of core materials, stringent limits may have to be placed on the Li content of those core materials which are exposed to significant neutron fluence which will increase their costs unnecessarily, or the capacity of the He purification system will have to be increased.

4. <u>SCHEDULE REQUIREMENTS</u>

Final data by 9/90 (start of final design phase).

5. <u>PRIORITY</u>

Urgency: 2
Cost benefit: L
Uncertainty in existing data: H
Importance of new data: L

6. <u>FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION</u>

Alternative 1 with the attendant conservative H^3 source terms as a result of not taking credit for major tritium sinks such as the core graphite. If no credit is taken for retention of H^3 by core materials, the limits on Li impurities in core materials may become more stringent or the capacity of the He purification system may have to be increased.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION PRODUCT DEPOSITION CHARACTERISTICS FOR STRUCTURAL METALS
DDN M.07.06
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Condensible fission products, including iodines and volatile fission metals, released from the core during normal operation and during accidents will tend to deposit in the primary circuit, thereby attenuating their release to the environment by orders of magnitude. On the other hand, this plateout activity is a major contributor to the occupational exposure during maintenance and ISI. In order to predict the amount and distribution of plateout in the NSSS, the deposition characteristics of condensible radionuclides on structural metals must be quantified.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.2 "Control Transport in Primary Circuit," Assumption 4: The available data, design methods, and computer codes for predicting transport in the primary circuit are accurate to within a factor of 10x at 95% confidence.

F2.1.4.1.1.2.2 "Protect Capability to Control Transport in Primary Circuit," Assumption 5: Design methods and codes for predicting fission product transport under core conduction cooldown conditions are accurate to within [10x] at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 2B: Methods for predicting fission product transport in the primary circuit under core conduction cooldown conditions will be validated sufficiently to assure an uncertainty factor of $\leq$ 10x at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 3C: Methods for predicting radionuclide transport in the primary and secondary coolant circuits [including the effects of H_2O] will be validated sufficiently to assure an uncertainty factor of $\leq$ 10x at 95% confidence.

1.2 Current Data Base Summary

The reference correlations which describe the deposition behavior of condensible radionuclides on structural metals have very large uncertainties (>10x). A major cause of these large uncertainties is that the sorption isotherms were typically measured in the laboratory at partial pressures orders of magnitude higher than those which occur in the reactor; moreover, for Cs, and Ag, the isotherms used for reactor design were measured on nonreference materials. The effects of surface films, dust, and particularly H_2O on plateout are highly uncertain, as essentially no quantitative data are available.

The current data base is inadequate to determine whether or the diffusion of deposited fission products into the interior of structural metals ("indiffusion") must be modelled under Modular HTGR operating conditions. There are FRG data, largely on nonreference materials, which imply that indiffusion must be modelled for surface temperatures above about 600°C.

1.3 Data Needed

Data are needed to characterize the deposition of I, Cs, and Ag on primary circuit structural metals. Correlations are needed which give the sorptivities of these nuclides as a function of temperature, partial pressure, surface state, and coolant chemistry for normal operating conditions, for H_2O plus pressure relief transients, and for core conduction cooldown transients; these sorption data should be obtained at representative partial pressures to avoid the orders-of-magnitude extrapolations which are necessary with the present data base. Particular attention should be given to the effects of dust (see DDN M07.09) and H_2O (see DDN M07.08) on the deposition process and to the possibility of chemical reactions involving fission products under core conduction cooldown conditions (e.g., CsI formation).

The diffusivities of the above nuclides in primary-circuit metals are needed under steady-state operating conditions, with special attention to the effects of surface films, in order to determine whether or not indiffusion must be explicitly modelled under Modular HTGR operating conditions. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Coolant Impurity Levels (Normal Operation)	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <6.3 x 10^5 μatm 630 μatm H_2
Coolant Pressure	1 atm

FP Partial Pressure	$<<10^{-10}$ atm
Primary Circuit Materials	Alloy 800H, 2-1/4 Cr 1 Mo, SA 533 (for I only)
Fission Products of Interest	I, Cs > Ag

Water Ingress Plus Depressurization

Environment	He/H_2O
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels (Transient)	[0.01 - 1.0] atm H_2O
Coolant Pressure	1 atm
Steam Quality	0 to 100%
FP Partial Pressure	$<<10^{-10}$ atm

Core Conduction Cooldown Transients

Environment	He; $He/H_2O/CO/H_2$; CO/N_2
Primary Coolant Temperature Range	[TBD] °C
Pressure	1 atm
Reynolds Number	[TBD]
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O [TBD] atm CO [TBD] atm H_2
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
FP Partial Pressure	[TBD] atm
Fission Products of Interest	I > Cs, Ag

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Accept the large uncertainties in the existing plateout correlations.

2. Use FRG plateout correlations derived from data on German materials as available and assume applicability to U.S. materials of construction.

3. Do not take credit for plateout as a fission product removal mechanism during normal operation and accidents.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure plateout characteristics of key nuclides under conditions representative of normal operation, H_2O ingress plus pressure relief transients and core conduction cooldown transients. Correlate the data for use in reactor design and safety analysis. Failure to take credit for plateout as a removal mechanism will impose exceedingly stringent requirements on the other barriers to fission product release to the environment, especially the SiC coating of the TRISO particle.

4. SCHEDULE REQUIREMENTS

Preliminary data by 3/89 (6 months prior to PSSAR); final data by 9/92 (1 year prior to FSSAR).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The first alternative would be the fallback approach, with sufficient conservatism added to the fission product source terms to account for the large uncertainties in plateout predictions. The risk is that with these very large uncertainties, the NRC will not allow any credit for plateout as a removal mechanism in assessing normal and accident doses. This eventuality would impose stringent requirements on the fuel performance which would be difficult to assure with a high degree of confidence.

D L Hanson 3/19/87
Originator Date

R J Turner 3/20/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION PRODUCT REENTRAINMENT CHARACTERISTICS FOR STRUCTURAL METALS
DDN M.07.07
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Fission products which deposit in the primary circuit during normal operation may be partially reentrained and released from the NSSS during rapid depressurization transients. The potential for reentrainment, or "liftoff," is apparently increased if particulate matter ("dust") or friable surface films are present in the primary circuit. Consequently, the reentrainment characteristics of fission products deposited on structural metals must be quantified, including the effects of dust.

1.1 Summary of Function Number/Title/Assumptions

F2.1.4.1.1.2.2 "Protect Capability to Control Transport in Primary Circuit," Assumption 4: Adequate data and validated methods will be available to predict reentrainment and redeposition of fission products in the primary circuit to within a factor of [10x] at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 1C: Validated methods will be available to describe the reentrainment and redeposition of plateout activity in the primary circuit to within a factor of 10x at 95% confidence.

1.2 Current Data Base Summary

The correlations for predicting fission product reentrainment during dry and wet depressurization transients contain very large uncertainties (>>10x). With one exception, the liftoff data base was obtained in ex situ blowdown tests wherein the blowdown specimens were mechanically removed from the loop or reactor in which the plateout activity was originally deposited. These ex situ blowdown data scatter badly and have been shown to be nonreproducible. The fractional liftoff of deposited activity was observed to be a function of the shear ratio - the ratio of the wall shear stress during the blowdown to that during normal operation - and the duration of the blowdown; no correlation between the fractional liftoff and the blowdown temperature or the humidity of the blowdown helium was evident although such a dependence could have been obscured by the excessive scatter in the data. Moreover, the effects of dust on liftoff have not been quantified.

In the single in situ blowdown test of the CPL 2/4 in-pile loop, <0.5% liftoff of the plateout activity resulted; however, this in situ loop blowdown was an integral validation test and is therefore unsuitable for defining the functional dependences between the fractional liftoff and system parameters such as wall shear stress, temperature, etc.

1.3 Data Needed

The extent to which plated activity may be removed during rapid depressurization transients must be quantified, including the effects of dust. Correlations are required which give the fractional liftoff of the radiologically important radionuclides I, Sr, and Cs as a function of the controlling system parameters. Test variables which must be investigated include shear ratio, absolute wall shear stress, blowdown duration, temperature, humidity, and surface oxidation state (other influential parameters may be identified in course of the testing program). The effects of high moisture levels are addressed under DDN M.07.08. The effects of dust on the reentrainment characteristics of deposited activity must be quantified (see DDN M.07.09 for detailed definition of data needed). Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are required for normal operating and transient conditions listed below.

Normal Operation (Initial Conditions prior to Blowdown)

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 atm
Reynolds Number	> 5000
Primary Circuit Materials	Alloy 800H, 2 1/4 Cr 1 Mo, SA 533
Metal Temperature Range	
Alloy 800H	400 - 700°C
2-1/4 Cr 1 Mo	200 - 450°C
SA 533 (for I only)	300 - 400°C
Fission Products of Interest	I, Sr > Cs

Rapid Depressurization

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels (Transient)	[126 - TBD] µatm H_2O [315] µatm CO [126] µatm CO_2 Total Oxidants <[630 - TBD] µatm [630] µatm H_2
Coolant Pressure	> 10 to 1 atm
Reynolds Number	> 5000
Shear Ratio*	0.5 to 5
Blowdown Duration	[1 - 10] min

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Develop alternate high-quality fuel with contamination and coating defect fractions of $<10^{-6}$ and accept sufficiently stringent tech specs on primary circuit activity so that 100% liftoff can be tolerated.

2. Rely on currently available liftoff data. Argue that only the liftoff data from the integral, in-pile, in situ blowdown test (CPL 2/4 test) are relevant.

3. Argue that rapid depressurization accidents with shear ratios greater than unity are incredible and that, on physical grounds, liftoff must be negligible for shear ratios less than unity.

4. Add a PWR-type containment building to the MHTGR design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure liftoff characteristics of key radionuclides under conditions representative of rapid depressurization transients. Correlate the data for use in reactor design and safety analysis. Failure to take credit for limited fission product liftoff during rapid depressurization transients would impose exceedingly stringent requirements on the other barriers to fission product release from the NSSS, especially the SiC coating of the TRISO particle.

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*Shear ratio is the ratio of the wall shear stress during the transient to the wall shear stress during normal operation.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (one year prior to PSSAR); final data by 9/92 (one year prior to FSSAR).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to add a PWR-type containment to the 350 MW(t) Modular HTGR design which would assure acceptable offsite doses during rapid depressurization transients; however, the extent to which the containment building would become contaminated by liftoff during a rapid depressurization transient and the consequent impact on investment risk (Goal 2 considerations) would have to be assessed. The consequences of adding containment are a $40-50M capital cost penalty per plant and possible design and licensing issues regarding the integrity and reliability of high-pressure containment buildings when used with HTGRs.

DL Hanson 3/19/87
Originator Date

RF Turner 3/20/87
Department Manager Date

[illegible] for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION PRODUCT WASHOFF CHARACTERISTICS FOR STRUCTURAL METALS
DDN M.07.08
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Fission products which deposit in the primary circuit during normal operation may be partially removed and subsequently released from the NSSS during combined H_2O ingress and depressurization transients. Consequently, the washoff characteristics of fission products deposited on structural metals must be quantified, including the effects of water chemistry.

1.1 Summary of Function Number/Title/Assumptions

F2.1.4.1.1.2.2 "Protect the Capability to Control Transport in Primary Circuit," Assumption 4: The available data, design methods, and computer codes for predicting transport in the primary circuit are accurate to within a factor of [10] at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 3C: Methods for predicting radionuclide transport in the primary and secondary coolant circuits [including the effects of H_2O] will be validated sufficiently to assure an uncertainty factor of $\leq$ [10] at 95% confidence.

1.2 Current Data Base Summary

There are no direct measurements of the washoff characteristics of key radionuclides deposited on primary circuit metals to permit a rigorous evaluation of the extent of fission product washoff during steam ingress accidents. Some LWR data on the behavior of fission products in steam-water systems may be relevant to HTGRs. The Germans have reportedly investigated the effects of water ingress on Cs plateout in the SMOC loop, but the data are not currently available to the U.S. program.

1.3 Data Needed

The extent to which plated out activity may be removed during water ingress events must be quantified, including the effects of dust. Correlations are required which give the fractional washoff of the radiologically important radionuclides I, Sr, and Cs as a function of the controlling system parameters. Test variables which must be investigated include temperature, pH, contact time, steam quality, Reynolds Number, and surface oxidation state (other influential parameters may be identified in course of the testing program). The effects of dust on the characteristics of deposited activity must be quantified (see DDN M07.09 for detailed definition of data needed). Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are required for normal operating and transient conditions listed below.

Normal Operation (Initial Conditions prior to H_2O Ingress)

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 atm
Reynolds Number	> 5000
Primary Circuit Materials	Alloy 800H, 2 1/4 Cr 1 Mo, SA 533 (for I only)
Metal Temperature Range	
Alloy 800H	400 - 700°C
2-1/4 Cr 1 Mo	200 - 450°C
SA 533 (for I only)	300 - 400°C
Fission Products of Interest	I, Sr > Cs

Water Ingress

Environment	He/H_2O
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels	[0.01 - 1] atm H_2O
Coolant Pressure	> 10 to 1 atm
Metal Temperature Range	[TBD]
Reynolds Number	> [TBD]

Shear Ratio*	≤ 1.0
Steam Quality	[1 to 100]%
pH Range	4 - 10
Contact Time	[0.1 - 10] h

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Develop alternate high quality fuel with contamination and coating defect fractions of $<10^{-6}$ and accept sufficiently stringent tech specs on primary circuit activity so that 100% washoff can be tolerated.

2. Use LWR data on the partitioning of fission products in steam-water systems.

3. Assume that any plateout activity washed off during steam ingress would stay in the liquid phase which would be largely retained within the primary circuit.

4. Design the NSSS to accommodate the maximum credible H_2O ingress without pressure relief and argue that the probabilities of any other combined H_2O ingress plus depressurization scenarios are $<5 \times 10^{-7}$/yr.

5. Add a PWR-type containment building to the MHTGR design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure washoff characteristics of key radionuclides under conditions representative of water ingress events. Correlate the data for use in reactor design and safety analysis. Failure to take credit for limited fission product washoff during H_2O ingress plus depressurization transients would impose exceedingly stringent requirements on the other barriers to fission product release from the NSSS, especially the SiC coating of the TRISO particle.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (one year prior to PSSAR); final data by 9/92 (one year prior to FSSAR).

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*Shear ratio is the ratio of the wall shear stress during the transient to the wall shear stress during normal operation.

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is a combination of Alternatives 2 and 3, viz., to argue on the basis of the LWR data that the dissolved radionuclides will remain in the liquid phase and that the liquid water will be largely retained within the NSSS. The risk is that without any direct measurements, the NRC will assume a large fractional washoff in assessing the offsite doses resulting from H_2O ingress plus depressurization accidents. This eventuality would impose stringent requirements on the fuel performance which would be difficult to assure with a high degree of confidence. The ultimate result could well be the necessity of incorporating a PWR-type containment into the design with an associated capital cost penalty of $40-50M per plant.

DL Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

CHARACTERIZATION OF THE EFFECT OF DUST ON FISSION PRODUCT TRANSPORT
DDN M.07.09
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The presence of circulating and/or deposited particulate matter in the primary circuit of an HTGR may alter the plateout distributions in the primary circuit during normal operation and may increase the extent to which condensible radionuclides are released from the primary circuit during dry and wet depressurization transients. Consequently, the effects of dust on the transport of condensible fission products in the primary coolant circuit must be characterized.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.2 "Control Transport in Primary Circuit," Assumption 4: The design methods and codes for predicting transport in the primary circuit will be shown to be accurate to within a factor of 10x at 95% confidence.

F2.1.4.1.1.2.2 "Protect Capability to Control Transport in Primary Circuit," Assumption 4: Validated methods will be available to predict reentrainment and redeposition of fission products in the primary circuit to within a factor of [10x] at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 1C: Validated methods will be available to describe the reentrainment and redeposition of plateout activity in the primary circuit to within a factor of 10x at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 2B: Methods for predicting fission product transport in the primary circuit under core conduction cooldown conditions will be validated sufficiently to assure an uncertainty factor of $\leq$ 10x at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 3C: Methods for predicting radionuclide transport in the primary and secondary coolant circuits [including the effects of H_2O] will be validated sufficiently to assure an uncertainty factor of $\leq$10x at 95% confidence.

1.2 Current Data Base Summary

The available data on the effects of dust on fission product transport in the primary coolant circuit are largely from reactor surveillance measurements made at Peach Bottom and AVR. However, the particulate matter in the primary circuits of these two reactors is carbonaceous so the relevance of these data is questionable for the 350 MW(t) Modular HTGR wherein a metal-oxide aerosol is more likely.

There are also British data on the transport of metal-oxide aerosols in AGRs, but no data on the effects of such aerosols on fission product transport.

Limited data are also available from the GA deposition loop program. In one test, a quantity of graphite powder was added to the out-of-pile loop, and the result was to alter the plateout distribution of the Cs-137 and Sr-90 and to increase significantly (>10x) the amount of liftoff observed in ex situ blowdown tests.

Finally, there is an extensive amount of open-literature data related to aerosol formation, transport, deposition, and reentrainment, but none relates directly to the circumstances expected in the primary circuit of the 350 MW(t) Modular HTGR.

1.3 Data Needed

Measurements under representative 4 x 350 MW(t) Modular HTGR conditions which elucidate the effects of particulate matter ("dust") on the transport of condensible radionuclides in the primary coolant circuit during normal operation and during transients, especially the effects upon the reentrainment/redeposition characteristics during dry and wet depressurization transients. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are needed to characterize the effect of dust on the transport and reentrainment of key fission product nuclides in the primary circuit under normal and transient conditions. The service conditions of interest are listed below.

Normal Operation

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C

Coolant Impurity Levels (Normal Operation)	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 atm
Reynolds Number	> 5000
Primary Circuit Materials	Alloy 800H, 2-1/4 Cr 1 Mo,
Metal Temperature Range	
Alloy 800H	400 - 700°C
2-1/4 Cr 1 Mo	200 - 450°C
Particulate Matter	
Composition	Ferritic metal oxide, graphite
Particle Size Distribution	[0.1 - 10 x 10^{-6}] m
Gasborne Concentration	[3 x 10^{-3}] g/m³
Surface Loading	[5] g/m²
Fission Products of Interest	Sr, I > Cs

<u>Rapid Depressurization</u>

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels	[126 - TBD] μatm H_2O [315] μatm CO [126] μatm CO_2 Total Oxidants <[630 - TBD] μatm [630] μatm H_2
Coolant Pressure Range	> 10 to 1 atm
Reynolds Number	> 5000
Shear Ratio*	0.5 to 5

*Shear ratio is the ratio of the wall shear stress during the transient to the wall shear stress during normal operation.

Water Ingress

Environment	He/H_2O
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels (Transient)	[0.01 - 10] atm H_2O,
Coolant Pressure Range	> 10 to 1 atm
Reynolds Number	> [TBD]
Shear Ratio*	$\leq$ 1
Steam Quality	[0 to 100]%
Contact Time	[0.1 - 10] h

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use current design assumptions and associated uncertainties which do not explicitly account for the effects of dust.

2. Characterize dust and dust-borne fission product behavior by analysis of existing data from Peach Bottom, AVR, and FSV.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measurements under representative Modular HTGR conditions which elucidate the effects of particulate matter ("dust") on the transport and deposition of condensible radionuclides in the primary coolant circuit during normal operation and, especially, the effects upon the reentrainment/redeposition characteristics during dry and wet depressurization transients.

A key technical issue in justifying the use of a low-pressure reactor building rather than a PWR-type containment building for the 4 x 350 MW(t) Modular HTGR is to demonstrate acceptable offsite doses during rapid depressurization accidents. The dominant source of radionuclide release under these circumstances is the reentrainment, or "liftoff" of plateout

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* Shear ratio is the ratio of the wall shear stress during the transient to the wall shear stress during normal operation.

activity in the primary circuit. The available data suggest that the presence of particulate matter ("dust") in the primary circuit may increase significantly (>10x) the potential for fission product liftoff. The current data base is inadequate to quantify, or even to reasonably bound, the effects of dust on radionuclide transport.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/89 (end of preliminary design); final data by 9/92 (one year prior to FSSAR submittal).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is the first alternative with tighter limits on the allowable plateout activity in the primary circuit to account for the uncertainties in the effects of dust on liftoff. Failure to better quantify the effects of dust on fission product liftoff could result in licensing delays, stringent tech specs on primary circuit activity, or the necessity of adding a PWR-type containment building to the Modular HTGR during the final design phase.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

[illegible] for G. C. Bramblett 3.25.87
Manager, Project Operations Date

[40-36]

DATE: 3/27/87

TRITIUM PERMEATION OF STEAM GENERATOR TUBES
DDN M.07.10
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Tritium permeation through steam generator tubes is expected to be the dominant pathway for tritium release to the environment; consequently, H^3 permeation in reference steam generator tube materials needs to be quantified under the relevant environmental conditions, including the effects of thermal cycling as a result of reactor startup and shutdown.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.2 "Control Transport in Primary Circuit," Assumption 7: H^3 permeation of steam generator tubes is adequately described by reference correlations to within a factor of [TBD] at 95% confidence.

1.2 Current Data Base Summary

The present data base on tritium permeation is limited and shows a pressure dependence and a strong effect of surface films that are not understood. In laboratory tests the presence of oxide films dramatically reduces the H^3 permeation through steam generator tube specimens; however, it remains to be demonstrated that this same benefit is realized in an operating HTGR or whether the effect is mitigated by thermal cycling, etc.

There are FRG permeation data for nonreference steam generator tube materials. The U.S. fusion program has also generated considerable permeation data, some of which may be relevant to the MHTGR.

1.3 Data Needed

Correlations describing the permeation of tritium through Alloy 800H and T22 (2-1/4% Cr 1% Mo steel) as a function of temperature, H^3 partial pressure, system pressure, coolant impurity concentrations and tube surface state. The effects of thermal cycling, which would occur as a result of reactor startup, shutdown, and load following, also must be determined. Quality Assurance must be in accordance with the requirements Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are summarized below.

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
SG Tube Materials	Alloy 800H, 2-1/4 1 Mo
Metal Temperature Range	
Alloy 800H	400 to 700°C
2-1/4 Cr 1 Mo	200 to 450°C
Thermal Cycling	[TBD] °C
H^3 Concentration Range	[TBD] Ci/m^3
Coolant Impurity Levels	126 µatm H_2O 315 µatm CO 126 µatm CO_2 Total Oxidants <630 µatm 630 µatm H_2
Coolant Pressure Range	> [10] to 1 atm

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use reference correlations and accept the present stringent limits on He^3 impurity in primary He and on Li in core materials.

2. Use FRG tritium permeation data as available and assume applicable to U.S. materials of construction.

3. Scale the measured FSV tritium discharges to the environment and do not attempt to predict tritium transport (very conservative because of the frequent H_2O ingresses and the design of the FSV He purification system).

4. Increase the capacity of the He purification system.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure tritium permeation rates through Alloy 800H and 2-1/4% Cr 1% Mo steel) as function of specified variables and develop correlations that can be used in analytical models for calculating an overall tritium mass balance for the 4 x 350 MW(t) Modular HTGR. The use of the present tritium permeation data could significantly overestimate the permeation rates and result in an unnecessarily restrictive limit on circulating tritium in the primary circuit which could in turn result in excessively stringent (expensive) limits on the He^3 impurity in primary coolant helium and on Li impurities in core materials.

4. SCHEDULE REQUIREMENTS

Final data by 9/92 (1 year prior to FSSAR submittal).

5. PRIORITY

Urgency: 4
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The first alternative with possible refinements in the permeation rate correlations as a result of analysis of tritium discharge data from FSV and AVR. The likely consequences of nonexecution would be unnecessarily restrictive limits on circulating tritium and, in turn, on allowable He^3 impurity in primary helium and Li impurities in core materials; such limits will restrict the potential sources of supply and increase the cost of helium and raw materials.

DL Hanson 3/19/87
Originator Date

R7 Turner 3/20/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

FISSION PRODUCT TRANSPORT IN REACTOR BUILDING DURING CORE CONDUCTION COOLDOWN TRANSIENTS
DDN M.07.11
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The reactor building is a significant barrier to the release of radionuclides to the environment during core conduction cooldown transients; consequently, the natural removal mechanisms, including condensation, settling, and plateout which serve to attenuate radionuclide release by at least an order of magnitude under these conditions need to be characterized. The PAG dose limits can not be met at the EAB with conservative source terms for wet or dry core conduction cooldown accidents without taking credit for the reactor building as a barrier to radionuclide release.

1.1 Summary of Function/Title/Assumptions

F3.1.1.2.3 "Control Transport from Reactor Building," Assumption 4: Data are available to adequately describe fission product transport and plateout in the reactor building.

1.2 Current Data Base Summary

No direct measurements have been made of radionuclide removal from contaminated helium by condensation, settling, and plateout under the conditions expected in the MHTGR reactor building during a core conduction cooldown transient. There is an extensive existing LWR data base on the behavior of radionuclides in steam-liquid water mixtures, and several major experimental programs are in progress on the behavior of radionuclides in LWR containment buildings (e.g., the DEMONA tests in the FRG). These LWR data, especially those which relate to radionuclide transport in containment buildings, conditions, may be applicable to HTGR systems.

1.3 Data Needed

Correlations describing the transport behavior of condensible radionuclides in the reactor building under wet and dry core conduction cooldown conditions are needed. The effects of temperature, coolant chemistry, surface state, and aerosols must be treated explicitly. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) must also be determined with particular attention to the effects of coolant chemistry on composition. The extent to which LWR data on radionuclide transport, especially transport in containment buildings, are applicable to the MHTGR must be determined. Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Dry Core Conduction Cooldown Transients - Reactor Building

Environment	Air/He, Air/He/CO/H_2
Pressure	1 atm
Flow Rate	[TBD] kg/s
Temperature Range	30 - 360°C
Range of Gasborne Impurities	[TBD] μatm H_2O [TBD] μatm CO [TBD] μatm CO_2 Total Oxidants <[TBD] μatm [TBD] μatm H_2
Gasborne Aerosols	
Composition	[TBD]
Particle Size	[TBD] m
Concentration	[TBD] kg/m^3
Materials of Construction	Concrete, [TBD]
Surfaces	Painted, unpainted
Radionuclides of Interest	I, Sr > Cs > Ag, Te
Radionuclide Partial Pressures	
I	2×10^{-10} - 2×10^{-7} atm
Sr	3×10^{-14} - 5×10^{-12} atm
Cs	2×10^{-12} - 2×10^{-7} atm
Ag	[3×10^{-18}] - 5×10^{-13} atm
Te	1×10^{-13} - 2×10^{-10} atm

Wet Core Conduction Cooldown Transients - Reactor Building

Environment	Air/He/H_2O, Air/He/CO/H_2/H_2O
Pressure	1 atm
Flow Rate	[TBD] kg/s
Temperature Range	30 - 360°C

Range of Gasborne Impurities	[0.05] - 0.20 H_2O
	[TBD] vpm CO
	[TBD] vpm CO_2
	<[TBD] vpm total oxidants
	[TBD] vpm H_2
Gasborne Aerosols	
Composition	[TBD]
Particle Size	[TBD] m
Concentration	[TBD] kg/m^3
Materials of Construction	Concrete, [TBD]
Surfaces	Painted, unpainted
Radionuclides of Interest	I, Sr > Cs > Ag, Te
Radionuclide Partial Pressures	
I	$3 \times 10^{-9} - 2 \times 10^{-5}$ atm
Sr	$4 \times 10^{-12} - 1 \times 10^{-8}$ atm
Cs	$6 \times 10^{-9} - 9 \times 10^{-5}$ atm
Ag	$4 \times 10^{-13} - 4 \times 10^{-9}$ atm
Te	$2 \times 10^{-11} - 2 \times 10^{-9}$ atm

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Assume that the correlations used to predict plateout and settling in PWR containments under dry conditions and to predict condensation under wet conditions are applicable to the Modular HTGR.

2. Do not take credit for condensation, plateout, and settling in the reactor building during core conduction cooldown transients.

3. Develop alternate high quality fuel with contamination and coating defect fractions of $<<10^{-6}$ so that PAG limits can be met without taking credit for reactor building as a release barrier.

4. Add a PWR-type containment building to the MHTGR design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to obtain correlations describing the transport behavior of condensible radionuclides in the reactor building under core conduction cooldown conditions by measuring the sorptivities of I, Cs, Sr, and Ag on reactor building materials of construction. The effects of temperature, coolant chemistry, surface state, condensation, and aerosols will be determined. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) will be determined with particular attention to the effects of coolant chemistry on composition.

The first alternative is judged to be highly risky. The NRC may not accept the assertion that PWR data is applicable to the MHTGR without direct experimental confirmation, given the differences in fission product chemistry and reactor building environments for the two reactor types. The second alternative is not viable because the PAG dose limits can not be met at the EAB with conservative source terms for wet or dry core conduction cooldown transients without taking credit for the reactor building as a barrier to radionuclide release to the environment. The third alternative is rejected because commercial manufacture of fuel with $<10^{-6}$ defects is not economically viable. The fourth alternative is too expensive.

4. SCHEDULE REQUIREMENTS

Preliminary data, including a determination of the extent to which LWR data are applicable to the MHTGR are required by 9/88 (one year prior to PSSAR submittal); final data by 9/92 (one year prior to FSSAR submittal).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to argue that the correlations used to predict plateout and fallout in PWR containments are applicable to the Modular HTGR design. The NRC could reject this assertion and require that the EPZ for the MHTGR be set at a radius in excess of the EAB distance. In the worst case, the NRC could require the inclusion of a PWR-type containment in the MHTGR design; the consequences of adding containment are \$40-50M capital cost penalty per plant and possible design and licensing issues regarding the integrity and reliability of high-pressure containment buildings when used with HTGRs.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

for G. R. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

VALIDATION OF DESIGN METHODS FOR FISSION GAS RELEASE
DDN M.07.12
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes used to predict fission gas release from the core (SURVEY for normal operation and SORS for accidents), including the radiologically important radioiodines, must be validated to have the specified predictive accuracies for normal operating conditions, for H_2O ingress transients, and for core conduction cooldown transients.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1 "Control Transport from Core," Assumption 5: The existing design methods and computer codes for calculating fuel failure and fission gas release from prismatic cores are accurate to within 4x at 95% confidence.

F2.1.4.1.1.2.1 "Protect the Capability to Control Transport from Core," Assumption 2: The existing design methods and computer codes for calculating gas release, including iodine release, from a prismatic core during transients are accurate to within [4x] at 95% confidence.

F3.1.1.2.1 "Control Transport from Core," Assumption 5: Validated methods and data are available to adequately assess fuel failure, fission product release [from the core], and release from the nuclear steam supply system.

F3.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels, "Assumption 1: Validated methods are available to adequately determine release of fission gases under wet shutdown conditions.

1.2 Current Data Base Summary

The validity of the reference design methods for fission gas release during normal operation has been assessed by applying them to FSV, Peach Bottom, and several irradiation capsules. The noble gas release from FSV during the first three cycles of operation was overpredicted by about a factor of five; the cause of the over-prediction is ambiguous: fuel failure may have been overpredicted, or the long-term, in-pile effect of hydrolysis may be less severe than observed in lab tests, or a combination of both these effects. The noble gas release from Peach Bottom Core 2 at end-of-life was underpredicted by a factor of two or three; however, the dominant source of gas release was heavy-metal contamination so not all the features of the gas release methodology were tested. Both FSV and Peach Bottom Core 2 contained carbide fuel rather than the reference UCO/ThO_2 fuel. The fission gas release from irradiation capsules containing reference UCO/ThO_2 fuel is generally predicted to within a factor of about five. However, these capsules operated dry so the hydrolysis model was not tested.

The validity of the transient gas release model used to analyze core conduction cooldown transients has not been rigorously assessed.

1.3 Data Needed

An experimental data base is needed to validate the integrated models and core-survey codes used to predict fission gas release from the core during normal operation and under transient conditions in order to assure that the predictive methods are accurate to within [4x] at 95% confidence. Particular attention must be given to effects of hydrolysis during steady-state power operation and during wet shutdowns and to the transient release of iodines and noble gases under dry and wet core conduction cooldown conditions. The data for assessing the overall accuracy of the gas release methodology must be independent of the data from which the individual correlations in the overall design method were originally derived (fuel failure models, gas release models for contamination and failed particles, etc.). Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Fuel Operating Temperature	700 - 1250°C
Maximum Fissile Particle Burnup	22% FIMA
Maximum Fertile Particle Burnup	3% FIMA
Maximum Fast Fluence (E > 29 fJ)	5×10^{25} n/m^2
Primary Coolant Temperature Range	300 to 700°C
He Velocity (in Coolant Channel)	>10 m/s

Coolant Impurity Levels	126 µatm H_2 315 µatm CO 126 µatm CO_2 Total Oxidants <630 µatm 630 µatm H_2
Coolant Pressure	> 10 atm
Fission Products of Interest	I > Kr > Xe

<u>Wet Shutdown Conditions</u>

Environment	$He/H_2O/Air$
Fuel Temperature Range	[100 - 300] °C
Coolant Pressure Range	> [10] to 1 atm
Range of Coolant Impurity Levels	[0.01 - 1] atm H_2O [TBD] atm O_2 [TBD] atm N_2

<u>Core Conduction Cooldown Transients</u>

Environment	He; $He/CO/H_2$; CO/N_2
Fuel Temperature Range Pressurized Cooldown Depressurized Cooldown	 900 - 1200°C 1200 - 1800°C
Pressure	1 atm
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
Fission Products of Interest	I > Xe > Kr

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Rely upon existing in-pile data to provide validation fission gas release methods.

2. Rely upon comparisons of design codes with analytical solutions and other transport codes, including the FRG codes as available, through a series of benchmark calculations.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Obtain fission gas release data from operating reactors and/or in-pile experiments. Compare predicted and observed results and assess accuracy of design methods. Stringent limits on fission product release from the core have been specified for the Modular HTGR in order to meet PAG dose limits at the EAB without containment. With these very tight limits on core release, large design margins to compensate for the current uncertainties in the fission product transport methods can not be tolerated. Therefore, validation of the fission product transport methods is essential to avoid significant licensing delays and/or major retrofitting in the final design phase.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (one year prior to PSSAR); final data by 9/92 (one year prior to FSSAR).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The first alternative with the necessity of added conservatism in the design to compensate for calculational uncertainties. A weakened licensing position as a consequence of not being able to answer the obvious question of how well do the design methods predict the fission product transport behavior observed in operating reactors and in-pile tests. The ultimate consequence could be the necessity of adding PWR-type containment to assure the offsite dose limits are met at the required confidence level.

D L Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

VALIDATION OF DESIGN METHODS FOR FISSION METAL RELEASE
DDN M.07.13
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes used to predict the release of fission metals from the core (TRAFIC/COPAR and TRAMP/COPAR for normal operation and SORS for accidents) must be validated to have the specified predictive accuracies for normal operating conditions and for core conduction cooldown transients.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.1 "Control Transport from Core," Assumption 6: The existing design methods and computer codes for calculating fission metal release from a prismatic core are accurate to within 10x at 95% confidence.

F2.1.4.1.1.2.1 "Protect the Capability to Control Transport from Core," Assumption 3: The existing design methods and computer codes for calculating fission metal release from a prismatic core during core conduction cooldown transients, including the effects of redeposition in the colder portions of the core, are accurate to within a factor of [10] at 95% confidence.

F3.1.1.2.1 "Control Transport from Core," Assumption 5: Validated methods and data are available to adequately assess fuel failure, fission product transport and release from the nuclear steam supply system.

1.2 Current Data Base Summary

The validity of the methods for predicting fission metal release during normal operation have been assessed by applying them to predict the observed metal release in operating HTGRs (Peach Bottom Core 2 and FSV) and in irradiation capsules and in-pile loops (SSL1, SSL2, Idylle 03, the four CPL2 loops, and R2 K13). Most of the available data are for the Cs isotopes with a small amount of Ag and Sr data. In general, the releases of fission metals were underpredicted by factors of several and, in some cases, by more than an order of magnitude. The cause of the underpredictions is ambiguous because the SiC defect fractions and the particle failure fractions are typically not well known; however, there is strong circumstantial evidence suggesting that the transport across the fuel compact/fuel element gap and the transport in the graphite web are not properly modelled. Moreover, with the exception of the R2 K13 data, the available data were obtained on nonreference fuels.

The validity of the methods for predicting fission metal release during core conduction cooldown transients have not been assessed systematically.

1.3 Data Needed

An experimental data base is needed to validate the integrated models and core-survey codes used to predict fission metal release from the core during normal operation and under dry and wet core conduction cooldown conditions in order to assure that the predictive methods are accurate to within 10x at 95% confidence. Particular attention must be given to the effects of irradiation and environment on the transport of fission metals in core graphite. The data for assessing the overall accuracy of the metal release methodology must be independent of the data from which the individual correlations in the overall method were originally derived (fuel failure models, graphite diffusivities and sorptivities, etc.). Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Fuel Operating Temperature	700 - 1250°C
Maximum Fissile Particle Burnup	22% FIMA
Maximum Fertile Particle Burnup	3% FIMA
Graphite Operating Temperature	300 - 1100°C
Maximum Fast Fluence (E > 29 fJ)	5×10^{25} n/m^2
Primary Coolant Temperature Range	300 to 700°C
He Velocity (in Coolant Channel)	> 10 m/s
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2

Pressure	> 10 atm
Fission Products of Interest	Cs > Ag > Sr

<u>Core Conduction Cooldown Transients</u>

Environment	He; He/CO/H_2; CO/N_2
Fuel Temperature Range	
Pressurized Cooldown	900 - 1200°C
Depressurized Cooldown	1200 - 1800°C
Graphite Temperature Range	
Pressurized Cooldown	700 -1200°C
Depressurized Cooldown	1200 - 1800°C
Pressure	1 atm
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
Fission Products of Interest	Sr > Cs > Ag

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Rely upon the existing U.S. data base to provide validation of fission metal release methods.

2. Rely upon comparisons of design codes with analytical solutions and other transport codes, including the FRG codes as available, through a series of benchmark calculations.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Obtain fission metal release data from operating reactors or in-pile experiments. Compare predicted and observed results and assess accuracy of design methods. Stringent limits on fission product release from the core have been specified for the Modular HTGR in order to meet PAG dose limits at the EAB without containment. With these very tight limits on core release, large design margins to compensate for the current uncertainties in the fission product transport methods can not be tolerated. Therefore, validation of the fission product transport methods is essential to avoid licensing delays and/or major retrofitting in the final design phase.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (1 year prior to PSSAR submittal); final data by 9/92 (1 year prior to FSSAR submittal).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The first alternative with the necessity of added conservatism in the design to compensate for calculational uncertainties. A weakened licensing position as a consequence of not being able to answer the obvious question of how well do the design methods used to predict Modular HTGR source terms predict the fission product transport behavior observed in operating reactors and in-pile tests. The ultimate consequence could be the necessity of adding PWR-type containment to assure the offsite dose limits are met at a high confidence level.

DL Hanson 3/19/87
Originator Date

R F Turner 3/20/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

VALIDATION OF DESIGN METHODS FOR PLATEOUT DISTRIBUTION
DDN M.07.14
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes used to predict the plateout per pass and the plateout distributions of condensible radionuclides in the NSSS (PADLOC) must be validated to have the specified predictive accuracies for normal operating conditions, for H_2O ingress transients, and for core conduction cooldown transients.

1.1 Summary of Function Number/Title/Assumptions

F1.1.4.1.1.2.2 "Control Transport in Primary Circuit," Assumption 4: The available data, design methods and computer codes for predicting transport in the primary circuit are accurate to within a factor of 10x at 95% confidence.

F2.1.4.1.1.2.2 "Protect the Capability to Control Transport in Primary Circuit," Assumption 5: Design methods and codes for predicting fission product transport in primary circuit under core conduction cooldown conditions are accurate to within [10X] at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 2B: Methods for predicting fission product transport in the primary circuit under core conduction cooldown conditions will be validated sufficiently to assure an uncertainty factor of $\leq$ 10x at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 3C: Methods for predicting radionuclide transport in the primary coolant circuit [including the effects of H_2O] will be validated sufficiently to assure an uncertainty factor of $\leq$ 10x at 95% confidence.

1.2 Current Data Base Summary

The validity of the methods used to predict plateout distributions in the primary coolant circuit during normal operation have been assessed by applying them to predict the plateout distributions observed in operating HTGRs (Peach Bottom and Dragon), in in-pile loops (VAMPYR 01 and the four CPL2 loops), and in out-of-pile loops (GA deposition loop, LAMINAR and SMOC). The plateout distributions of Cs in Peach Bottom and of Cs, I, and Ag in Dragon were predicted to within a factor of two or three; however, most of these data are for plateout surface temperature in the range of 250°C to 500°C. At temperatures >500°C, the predicted surface concentrations, especially for iodines, are orders of magnitude lower than observed. The plateout distributions observed at the higher temperatures can not be explained by the reversible surface adsorption model which is the physical basis for the reference plateout methodology. The effects of dust on the plateout process have not been systematically investigated.

The validity of the methods used to predict plateout under core conduction cooldown conditions has not been systematically assessed.

1.3 Data Needed

An experimental data base is needed to validate the integrated models and codes used to predict fission product plateout in the primary circuit during normal operation and under core conduction cooldown conditions in order to assure that the predictive methods are accurate to within 10x at 95% confidence. Particular attention must be given to the effects of dust and environment on the transport in the primary circuit and to the necessity of modeling diffusion of plated out activity into the interior of structural materials. The data for assessing the overall accuracy of the plateout methodology must be independent of the data from which the individual correlations in the overall method were originally derived (mass transfer coefficients, graphite and metal diffusivities and sorptivities, etc.). Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The reactor operating or service conditions are given below.

Normal Operation

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Plateout Surface Temperatures	300 to 700°C
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 atm
Reynolds Number	> 5000
Primary Circuit Materials	Alloy 800H, 2-1/4 Cr 1 Mo

Fission Products of Interest	I, Cs > Ag

<u>Water Ingress</u>

Environment	He/H_2O
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels (Transient)	[0.01 - 1.0] atm H_2O
Coolant Pressure	> [10] to 1 atm
Reynolds Number	> [TBD]
Shear Ratio*	$\leq$1.0
Steam Quality	[0 to 100]%
Fission Products of Interest	I, Sr > Cs

<u>Core Conduction Cooldown Transients</u>

Environment	He; He/H_2O/CO/H_2; CO/N_2
Primary Coolant Temperature Range	[TBD] °C
Pressure	1 atm
Range of Coolant Impurity Levels (Pressurized Cooldown)	[0.01 - 1.0] atm H_2O [TBD] atm CO [TBD] atm H_2
Range of Coolant Impurity Levels (Depressurized Cooldown)	[0, 0.35] atm CO [0, 0.65] atm N_2
FP Partial Pressure	[TBD] atm
Fission Products of Interest	I, Sr > Cs

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Rely upon the existing U.S. data base to provide validation of the plateout methods.
2. Rely upon comparisons of design codes with analytical solutions and other transport codes, including the FRG codes as available, through a series of benchmark calculations.

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*Shear ratio is the ratio of the wall shear stress during the transient to the wall shear during normal operation.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to obtain a data base for validation of plateout methods under normal operating and accident conditions in order to assure that the predictive methods are sufficient to meet the specified accuracy requirements. Stringent limits on fission product release from the NSSS have been specified for the Modular HTGR in order to meet PAG dose limits at the EAB without containment. With these very tight limits large design margins to compensate for the current uncertainties in the fission product transport methods can not be tolerated. Therefore, validation of the fission product plateout methods is essential to avoid licensing delays and/or major retrofitting in the final design phase.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (one year prior to PSSAR submittal); final data by 9/92 (one year prior to FSSAR submittal).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 1, i.e., rely upon existing U.S. data base and add margin to account for the considerable uncertainties. The consequences are potential licensing delays and the possible imposition of stringent tech specs until an adequate U.S. data base can be generated at a substantial increase in development costs.

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Originator Date

R J Turner 3/20/87
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DATE: 3/27/87

VALIDATION OF DESIGN METHODS FOR FISSION PRODUCT LIFTOFF
DDN M.07.15
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes (POLO) used to predict the extent to which fission products plated out in the primary coolant circuit during normal operation may be lifted off during depressurization transients and released from the NSSS must be validated to have the specified predictive accuracies.

1.1 Summary of Function Number/Title/Assumptions:

F2.1.4.1.1.2.2 "Protect Capability to Control Transport in Primary Circuit," Assumption 4: Adequate data and validated methods will be available to predict reentrainment and redeposition of fission products in the primary circuit to within a factor of [10x] at 95% confidence.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 1C: Validated methods will be available to describe the reentrainment and redeposition of plateout activity in the primary circuit to within a factor of 10x at 95% confidence.

1.2 Current Data Base Summary:

The present data base for the validation of fission product liftoff methods is extremely limited and does not explicitly account for the effects of dust. In the single in situ blowdown test of the CPL 2/4 in-pile loop, <0.5% liftoff of the plateout activity was observed. However, the maximum shear ratio realized in the CPL 2/4 blowdown was only 1.08 so these data do not provide a comprehensive test of a candidate liftoff model. Moreover, the CPL 2/4 loop was known to contain an inordinate amount of metal oxide aerosol which was not characterized; consequently, the CPL 2/4 data are likely to be biased high.

Despite their limitations, the CPL 2/4 data do provide reason to believe that the release from the NSSS due to liftoff will be ≤1% for the licensing basis events involving rapid depressurization. However, the existence of questionable ex situ blowdown data showing much higher liftoff (see DDN M.07.07, Section 1.2) and no comprehensive independent data base to refute these results have led to a liftoff model with excessive uncertainty and perhaps excessive conservatism for use in the analysis of depressurization transients.

1.3 Data Needed:

An experimental data base is needed to validate the integrated models and codes used to predict fission product liftoff and release from the NSSS during rapid depressurization transients in order to assure that the predictive methods are accurate to within 10x at 95% confidence. Particular attention must be given to the effects of dust and surface state on liftoff. The data for assessing the overall accuracy of the liftoff methodology must be independent of the data from which the individual correlations in the overall method were originally derived (see DDN M.07.07). Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The primary data needed will be liftoff fractions under the conditions described below for fission product plated out under normal operating conditions.

a. Baseline data on blowdown and liftoff with dry, dust-free helium coolant is needed.

b. Additional data on liftoff during blowdown with helium including dust or particulate matter.

The service conditions of interest are summarized below.

Normal Operation (Initial Conditions prior to Blowdown)

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 atm
Reynolds Number	> 5000
Particulate Matter Composition	Ferritic metal oxide, graphite
Particle Size Distribution	[0.1 - 10] μm
Gasborne Concentration	[3×10^{-3}] g/m³
Surface Loading	[5] g/m²
Primary Circuit Materials	Alloy 800H, 2-1/4 Cr 1 Mo

Fission Products of Interest	I, Sr > Cs

<u>Rapid Depressurization</u>

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Coolant Impurity Levels (Transient)	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 to 1 atm
Reynolds Number	> 5000
Shear Ratio*	0.5 to 3.0
Blowdown Duration	[1 - 10] min

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Develop alternate high quality fuel with contamination and coating defect fractions of $<10^{-6}$ and accept sufficiently stringent tech specs on primary circuit activity so that 100% liftoff can be tolerated.

2. Rely on currently available liftoff data. Argue that only the liftoff data from the single, in-pile <u>in situ</u> blowdown test (CPL 2/4 test) are relevant.

3. Argue that rapid depressurization accidents with shear ratios greater than unity are incredible and that, on physical grounds, liftoff must be negligible for shear ratios less than unity.

4. Add a PWR-type containment building to the 4 x 350 MW(t) design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Selected approach is the validation of design model for fission product liftoff. Failure to take credit for limited fission product liftoff during rapid depressurization transients would impose exceedingly stringent requirements on the other barriers to fission product release, especially the SiC coating.

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* Shear ratio is the ratio of the wall shear stress during the transient to that during normal operation

Commercial manufacture of fuel with <10^{-6} defects is not economically viable. A single in situ blowdown test is unlikely to be accepted as definitive by the NRC. Analysis of the 250 MW(t) Pebble MRS demonstrated that rapid depressurization transients with shear ratios >1.0 are credible for modular reactor designs. Finally, addition of a containment building is a viable option, but it would add $40-50M in capital cost per plant.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (one year prior to PSSAR); final data by 9/92 (one year prior to FSSAR).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to add a PWR-type containment to the 350 MW(t) Modular HTGR design which would assure acceptable offsite doses during rapid depressurization transients; however, the extent to which the containment building would become contaminated by liftoff during a rapid depressurization transient and the consequent impact on investment risk (Goal 2 considerations) would have to be assessed. The consequences of adding containment are $40-50M capital cost penalty per plant and possible design and licensing issues regarding the integrity and reliability of high-pressure containment buildings when used with HTGRs.

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Originator Date

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VALIDATION OF DESIGN METHODS FOR FISSION PRODUCT WASHOFF
DDN M.07.16
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/Overall Plant

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes (POLO) used to predict the extent to which fission products plated out in the primary coolant circuit during normal operation may be washed off during H_2O ingress transients and released from the NSSS must be validated to have the specified predictive accuracies.

1.1 Summary of Function Number/Title/Assumptions:

F2.1.4.1.1.2.2 "Protect Capability to Control Transport in Primary Circuit," Assumption 4: Adequate data and validated methods will be available to predict reentrainment and redeposition of fission products in the primary circuit.

F3.1.1.2.2 "Control Transport in Primary Circuit," Assumption 3C: Methods for predicting radionuclide transport in the primary and secondary coolant circuits will be validated sufficiently to assure an uncertainty factor of $\leq$10 at 95% confidence.

1.2 Current Data Base Summary:

There are no integral measurements of fission product washoff during steam ingress accidents which are presently available to the U.S. program. Some LWR data on the behavior of fission products in steam-water systems may be relevant to HTGRs. The Germans have reportedly investigated the effects of water ingress on Cs plateout in the SMOC loop, but the data are not currently available to the U.S. program.

1.3 Data Needed:

An experimental data base is needed to validate the integrated models and codes used to predict fission product washoff and release from the NSSS during water ingress plus depressurization transients in order to assure that the predictive methods are accurate to within 10x at 95% confidence. Particular attention must be given to the effects of surface state, dust, and water chemistry on washoff. The data for assessing the overall accuracy of the washoff methodology must be independent of the data from which the individual correlations in the overall method were originally derived (see DDN M.07.08). Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The primary data needed will be washoff under the conditions described below for fission product plated out under normal operating conditions.

The service conditions of interest are summarized below.

Normal Operation (Initial Conditions prior to H_2O Ingress)

Environment	Helium
Primary Coolant Temperature Range	300 to 700°C
Coolant Impurity Levels	126 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants <630 μatm 630 μatm H_2
Coolant Pressure	> 10 atm
Reynolds Number	> 5000
Particulate Matter Composition	Ferritic metal oxide, graphite
Particle Size Distribution	[0.1 - 10] μm
Gasborne Concentration	[3×10^{-3}] g/m^3
Surface Loading	[5] g/m^2
Primary Circuit Materials	Alloy 800H, 2-1/4 Cr 1 Mo
Fission Products of Interest	I, Sr > Cs

Water Ingress

Environment	He/H_2O
Primary Coolant Temperature Range	300 to 700°C
Range of Coolant Impurity Levels	[0.01 - 1.0] atm H_2O
Coolant Pressure	> 10 to 1 atm
Reynolds Number	> [TBD]
Shear Ratio*	$\leq$ [1]

*Shear ratio is the ratio of the wall shear stress during the transient to the wall shear during normal operation.

Steam Quality	[1 to 100] %
pH Range	>[7]
Contact Time	[0.1 - 10] h

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Develop alternate high quality fuel with contamination and coating defect fractions of $< 10^{-6}$ and accept sufficiently stringent tech specs on primary circuit activity so that 100% washoff can be tolerated.

2. Use LWR data on the partitioning of fission products in steam-water systems.

3. Assume that any plateout activity washed off during steam ingress would stay in the liquid phase which would be largely retained within the primary circuit.

4. Design the NSSS to accommodate the maximum credible H_2O ingress without pressure relief and argue that the probabilities of any other combined H_2O ingress plus depressurization scenarios are $<5 \times 10^{-7}$/yr.

5. Add a PWR-type containment building to the 4 x 350 MW(t) design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Selected approach is the validation of design model for fission product washoff. The commercial manufacture of fuel with contamination and coating defects $<10^{-6}$ is not economically viable. The applicability of LWR data to HTGR systems must be demonstrated. The assumption that the dissolved radionuclides would stay in the liquid phase is reasonable but requires experimental confirmation. The designing of NSSS without pressure relief would violate the ASME code for pressure vessels. Addition of a containment building is a viable alternative but would add $40-50M in capital cost per plant.

4. SCHEDULE REQUIREMENTS

Preliminary data by 9/88 (one year prior to PSSAR submittal); final data by 9/92 (one year prior to FSSAR submittal).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is a combination of Alternatives 2 and 3, viz., to argue on the basis of the LWR data that the dissolved radionuclides will remain in the liquid phase and that the liquid water will be largely retained within the NSSS. The risk is that without any direct measurements, the NRC will assume a large fractional washoff in assessing the offsite doses resulting from H_2O ingress plus depressurization accidents. This eventuality would impose stringent requirements on the fuel performance which would be difficult to assure with a high degree of confidence. The ultimate result could well be the necessity of incorporating a PWR-type containment into the design with an associated capital cost penalty of $40-50M per plant.

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UF_6-UO_3 CONVERSION PROCESS DEVELOPMENT
DDN M.07.17
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The preparation of a suitable feed material for the fissile kernel manufacturing process includes a process for converting UF_6 to UO_3. A 19.9% enriched UO_3 feed material is needed for the UCO kernel production process. The oxide feed material at 19.9% enrichment is not currently commercially available in the proper enrichment and form. Blending commercially available high and low enriched oxide to obtain 19.9% enrichment is wasteful of the highly enriched uranium. Production of 19.9% enriched UO_3 from 19.9% enriched UF_6 would provide the most cost effective long-term source of feed material for UCO kernel fabrication.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 5: Processes are available for manufacturing oxide-based fuel kernels.

1.2 Current Data Base Summary

Process Technology exists for commercial conversion of 3% and 93% enriched UF_6 to UO_2. It is anticipated that one of those processes could be modified to provide 19.9% enriched UNH to UO_3. However, at present no facilities exists for converting UF_6 to UNH or UO_3 or any other starting material for LEU (20%) kernel preparation.

1.3 Data Needed

Procedures, process parameters, equipment, and product specifications are needed for 19.9% enriched UF_6 conversion to either UNH or UO_3. In addition, an economic evaluation showing the capital requirements and manufacturing cost of LEU UO_3 related to production of the first core is needed to support fuel cycle cost analysis. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

The demonstrated procedures for LEU UF_6 conversion to UO_3 must satisfy the following parameters:

Solubility of UO_2 in acid	>99.5 wt % into solution
Acidity of final solution	$NO_3/U \leq 1.8$ (Molar Ratio)
Throughput (combination of modules)	Consistent with supplying feed stock for the UCO kernel line (~10 kg H.M./day)

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above-described data are:

2.1 Obtain 3% to 8% enriched U in UO_2 from existing sources, dilute with 93% enriched UO_2, and then convert the UO_2 into UO_3.

2.2 Adopt an LEU cycle with $\leq$10% enriched U. Obtain $\leq$10% enriched UO_2 from existing sources and then convert the UO_2 into UO_3.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Develop and demonstrate pilot line equipment to convert 19.9% enriched UF_6 to either UNH or UO_3. Also document process that provides acceptable feedstock for UCO kernel manufacture. This approach was selected because it provides an assured high quality feed material tailored to the needs of the UCO process. Alternate approaches 2.1 and 2.2 were considered too costly.

4. SCHEDULE REQUIREMENTS

A prototype process should be developed to assure that the appropriate, low-cost process is available for the start of production for the first core (9/92).

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is Alternative 2.1 which is to blend ≤10% enriched UO_2 with 93% enriched UO_2 to obtain 19.9% enrichment and adjust the UCO kernel process to accommodate the available raw material. The consequence of this approach would be a cost increase of about $7M for the first core, potential problems in kernel process development, and a less reliable source of suitable feedstock.

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UCO FISSILE KERNEL PROCESS DEVELOPMENT
DDN M.07.18
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The feasibility of producing 350 μm diameter UCO by the gelation/precipitation process has been demonstrated on a laboratory scale. The process must be scaled up and demonstrated at a pilot line size large enough to be certain scaling effects are properly treated and the process details, including costs, are accurately known for transition to commercial status.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 5: Processes are available for manufacturing oxide-based fuel kernels.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 7: Processes are available for manufacturing high-integrity coated fuel particles.

1.2 Current Data Base Summary

A process to manufacture UCO kernels has been developed and some product has been irradiation-tested. Parts of the demonstration unit are in place and its process parameters are developed.

1.3 Data Needed

A final demonstration of the UCO kernel production process must be made producing kernels with density, size, and chemical composition, complying with the fuel product specifications. In addition, the production rate and yield must be consistent with economic goals. Product quality must be established within the required 95% confidence level. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

Economic evaluation is also needed showing the capital requirements and manufacturing cost of UCO kernels related to production of MHTGR fuel.

1.4 Data Parameters/Service Conditions

The UCO kernel production process must demonstrate production of kernels with the following parameters:

Diameter	350 μm
Density	$\geq$ 10.5 Mg/m^3
Composition	$UC_{0.3}O_{1.7}$
Throughput	$\geq$ 5 kg/H.M./day-module
Enrichment	$\leq$19.9% U-235
Yield	$\geq$90%
Feed stock	UO_3

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1 Change reference kernel to UO_2 and purchase from NUKEM in the FRG.

2.2 Change reference kernel to UC_2 or $(U,Th)C_2$ and use VSM process at GA Technologies.

3. SELECTED APPROACH AND EXPLANATION

Build, demonstrate, and specify the process and equipment to prepare UCO kernels for accelerated irradiation testing and real time irradiation qualification test. This approach provides UCO kernels on schedule and with a minimum amount of risk. Alternative 2.1 was not selected because it does not provide a reliable, secure, domestic source of nuclear fuel. Secondly, foreign fuel may or may not comply with NRC regulations.

Alternative 2.2 was not selected because UC_2 kernels presently made via VSM process have a maximum diameter of 200 μm versus the desired diameter of 350 μm, and significant additional development would be needed to make the larger kernels.

4. SCHEDULE REQUIREMENTS

A prototype process should be in place to assure that the appropriate kernels are available to manufacture coated particles for the irradiation-proof test prior to the submission of the PSSAR (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position would be to change the kernel design so that existing UO_2 processes could be utilized. The consequence of this action would be reduction in performance margin, and if a foreign supplier (NUKEM) were used there would be increased cost and schedule risk associated with competing with foreign interests and priorities.

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DATE: 3/11/87

FUEL PARTICLE COATING PROCESS DEVELOPMENT
DDN M.07.19
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The lack of a containment building on the 4 x 350 MW(t) HTGR makes it necessary to reduce the coating defect, contamination, and in-pile failure relative to prior requirements. Without coating process improvements it will not be possible to produce fuel with the required low level of defects and contamination.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 7: Processes are available for depositing high-integrity TRISO coatings on oxide-based fuel kernels.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 7: Processes are available for manufacturing high-integrity coated fuel particles.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 5: Processes are available for manufacturing high-integrity coated fuel particles.

F3.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 4: Processes are available for depositing high-integrity coatings on oxide based fuel kernels.

1.2 Current Data Base Summary

A full-scale coater is in place and there is considerable experience in the U.S. in making TRISO-coated particles. However, extended campaigns of TRISO particle production under constant conditions have not been carried out, so uniformity of MHTGR quality fuel has not been demonstrated. Coating of 10 kg batches of ThO_2 is possible with the existing coater configuration, but the coater needs modification to permit the coating of $\geq$5 kg of 20% U in a critically safe condition. There is a large uncertainty in the fabrication cost of MHTGR quality TRISO fuel particles.

1.3 Data Needed

The coater must be modified to permit $\geq$5 kg of 20% U to be coated to demonstrate production of high quality fissile fuel at a production rate consistent with 4 x 350 MW(t) HTGR requirements. This will insure that scaling effects are properly accounted for in fissile particle coating. Documented process and process parameters capable of delivering particles within the desired product specifications. Manufacturing cost of TRISO coating must be established to support fuel cycle cost estimates. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Processes for fuel particle coatings must be capable of providing particles that satisfy the following targets:

Throughput	
Fissile	$\geq$ 10 kg/H.M./day (2 batches/day-coater)
Fertile	$\geq$ 20 kg/H.M./day (2 batches/day-coater)
Feed stock	Kernels from UCO/ThO_2 line

Design	ThO_2	UCO
Kernel dia. (μm)	500	350
Buffer thickness (μm)	65	100
IPyC thickness (μm)	50	50
SiC thickness (μm)	35	35
OPyC thickness (μm)	40	40

Quality Requirements	Fraction Fissile or Fertile	
	(50% Conf. on Mean)	(95% Conf. $\leq$5% of Fuel Rods Exceed)
Missing or Defective SiC	$\leq 5.0 \times 10^{-5}$	1.0×10^{-4}
Heavy Metal (HM) Contamination	$\leq 1.0 \times 10^{-5}$	2×10^{-4}
Total Fraction HM Outside Intact SiC	$\leq 6.0 \times 10^{-5}$	1.2×10^{-4}
Missing or Defective Inner PyC	$\leq 4.0 \times 10^{-5}$	1.0×10^{-4}
Missing or Defective Buffer Fraction	$\leq 5.0 \times 10^{-5}$	2.0×10^{-4}
Missing or Defective Outer PyC	$\leq 1.0 \times 10^{-4}$	1.0×10^{-3}

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1 Use Fort St. Vrain fuel particle coating technology.

2.2 License foreign TRISO coating processes for use in the U.S.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Modify the existing coater to be critically safe with $\geq$5 kg of 20% uranium. Complete the development with the modified coater, which can coat $\geq$5 kg H.M./batch for fissile and 10 kg H.M./batch for fertile, and produce the required high quality. Multiple batches each day and/or multiple modules will be needed to meet the heavy metal throughput goals. This will provide fuel particles for irradiation tests in a timely manner. Alternative 2.1 was not chosen because fission gas release from this type of fuel would be greater than allowed by the product specification. Alternative 2.2 was not chosen because it would require significant and new capital costs and delay the design and licensing schedule.

4. SCHEDULE REQUIREMENTS

A prototype process should be in place to assure that appropriate coated particles are available to manufacture fuel bodies for irradiation-proof tests and the work shall be completed at least 3 years prior to submission of the FSSAR (9/90).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position would be to use Fort St. Vrain fuel particle coating equipment and technology. The consequence of this action would be a risk that the quality of coating would not be adequate to meet the reference plant requirements. Extreme consequence would be the need of a containment building.

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DATE: 3/11/87

DEVELOPMENT OF PROCESSES FOR FUEL ROD COMPACT FABRICATION
DDN M.07.20
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel rod compact bonds the particles into an easily confined body with higher thermal conductivity than an unbonded particle bed. However, during fuel rod compact manufacture, particle coating damage and heavy metal contamination of the bonding matrix can take place. Because a containment building is not part of the 4 x 350 MW(t) HTGR design, fuel rod compacts must have improved quality compared to the 2240 MW(t) HTGR fuel rod compacts. For example, exposed heavy metal contamination levels are to be $\leq$4.0E-5 and the defective SiC coating fractions are to be <1.0E-4 at the 95% confidence level. In order to meet these more restrictive product specifications, substantial reductions in heavy metal contamination resulting from fuel particle coating breakage and cross contamination of the fuel rods during fabrication must be achieved.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high quality fuel compacts for inclusion in prismatic fuel elements.

F3.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high quality fuel compacts.

1.2 Current Data Base Summary

Extensive experience in the production of fuel rod compacts has been obtained at GA over many years. A fuel rod manufacturing production line for producing FSV fuel rods is currently operational at GA. However, product specifications for the FSV fuel rods are less restrictive than those required for the 4 x 350 MW(t) HTGR. Reduced defects and heavy metal contamination levels required for the 4 x 350 MW(t) HTGR design require improvements in the FSV fuel rod fabrication process. There is a large uncertainty in the fabrication cost for MHTGR quality fuel rods.

1.3 Data Needed

Process and equipment specifications for a demonstrated process to manufacture low defect fuel rod compacts to meet specifications for the 4 x 350 MW(t) HTGR design are needed. Documented process flowsheets, process specifications, and product quality data to support performance of fuel produced with this improved fuel rod manufacturing process are required. These data are essential in supporting performance for these fuels for plant design and licensing. Extended compact making campaigns ($\geq$24,000 compacts) are needed to establish variability and yield of the high quality fuel rod fabrication process. An economic evaluation of this unit operation is also required. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Processes for fuel compact making must be capable of providing compacts that satisfy the following goals:

Throughput Requirements	8000 compacts/day (from two 40-hole molds)
Dimensions	
- Diameter	12.5 mm
- Length	50 mm
Maximum Packing Fraction	0.58 (fuel + shim)

Quality Requirements	Fraction Fissile or Fertile	
	(50% Conf. on Mean)	(95% Conf. $\leq$5% of Fuel Rods Exceed)
Defective SiC	$\leq 5.0 \times 10^{-5}$	1.0×10^{-4}
Heavy Metal (HM) Contamination	$\leq 1.0 \times 10^{-5}$	2×10^{-5}
Total Fraction HM Outside Intact SiC	$\leq 6.0 \times 10^{-5}$	1.2×10^{-4}
Missing or Defective Inner PyC	$\leq 4.0 \times 10^{-5}$	1.0×10^{-4}

2. DESIGNER'S ALTERNATIVES

Alternatives to the selected approach are:

2.1 Develop and qualify a chemical cleaning process to remove exposed kernels from fabricated compacts containing UCO and ThO_2 particles.

2.2 Use the Fort St. Vrain fuel compact fabrication process in conjunction with a particle overcoating technique to minimize particle-to-particle interaction and failure during fuel rod particle compression and matrix injection.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The equipment, procedures, and specifications for making the fuel rod compacts will be adapted from the FSV-HTGR commercial process. The strategy is to make the transition from the FSV fuel particle to the thicker, more robust 4 x 350 MW(t) particle coating design and modify the control system for the injection press to avoid overpressure. It is anticipated that these changes will avoid failure from particle-to-particle interaction and the resulting contamination of the matrix with heavy metal. Alternatives 2.1 and 2.2 were not selected because they would entail as much or more development work with lower probability of technical success.

4. SCHEDULE REQUIREMENTS

A prototype process should be in place to provide fuel for the final irradiation-proof test (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

If this task is not carried out, the fallback position would be to adopt the FSV fuel compact process without modification, but with a chemical cleaning process after final heat treatment to reduce contamination and meet requirements. The risk of nonexecution of the selected approach is that as-manufactured fuel quality requirements would not be satisfied and the core would contain inferior quality fuel. The consequence would be a significant risk that technical specifications on primary circuit activity would not be satisfied which would result in excessive plant unavailability or the need for a containment building.

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Originator Date

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DATE: 3/11/87

QC TEST TECHNIQUES DEVELOPMENT
DDN M.07.21
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel produced for the 4 x 350 MW(t) HTGR will have extremely low contamination and defective particle fraction. The levels of contamination and defects will be below the detection limit for existing techniques employed in fuel production. Therefore, improved techniques for controlling quality of the fuel are needed.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high quality fuel compacts for inclusion in prismatic fuel elements.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 7: Processes are available for manufacturing high-integrity coated fuel particles.

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 5: Processes are available for manufacturing oxide-based fuel kernels.

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 7: Processes are available for depositing high-integrity TRISO coatings on oxide-based fuel kernels.

F3.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high-quality fuel.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 5: Processes are available for manufacturing high-integrity coated fuel particles and compacts.

F3.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 4: Processes are available for depositing high-integrity TRISO coatings on oxide based fuel kernels.

1.2 Current Data Base Summary

At present there are techniques and instruments to insure consistent, good quality of fuel for HTGR reactors. Those techniques were developed for FSV carbide type fuel. However, the 4 x 350 MW(t) modular reactor has oxide-based fuel with higher standards and present techniques fall short in several analyses. Namely, measurement of heavy metal contamination of the matrix, heavy metal dispersion in buffer, and missing buffer fraction.

1.3 Data Needed

Qualified and documented procedures for characterizing the fuel are needed to ensure compliance with fuel product specification requirements. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The qualified procedures for fuel characterization must be capable of ensuring the following parameters are met:

Quality Requirements	Fraction Fissile or Fertile	
	(50% Conf. on Mean)	(95% Conf. ≤5% of Fuel Rods Exceed)
Missing or Defective SiC	$\leq 5.0 \times 10^{-5}$	1.0×10^{-4}
Heavy Metal (HM) Contamination	$\leq 1.0 \times 10^{-5}$	2×10^{-5}
Total Fraction HM Outside Intact SiC	$\leq 6.0 \times 10^{-5}$	1.2×10^{-4}
Missing or Defective Inner PyC	$\leq 4.0 \times 10^{-5}$	1.0×10^{-4}
Missing or Defective Buffer Fraction	$\leq 5.0 \times 10^{-5}$	2.0×10^{-4}

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1 Use techniques developed for Fort St. Vrain fuel.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Procure or build satisfactory equipment as required and demonstrate QC methods for measuring heavy metal contamination in the fuel compact matrix, heavy metal dispersion in buffer, and missing buffers. Prepare QC test techniques, operating procedures, and test equipment documentation. Inspect fuel for irradiation testing using improved techniques as part of the process of qualifying the new techniques. To the extent possible, procedures developed at NUKEM will be adopted to minimize the development cost.

Alternative 2.1 was not selected because it would not yield the precision needed to detect the low level of contamination and defects specified for the MHTGR.

4. SCHEDULE REQUIREMENTS

A prototype process should be in place to assure that the final procedures are utilized to characterize the fuel irradiated in the accelerated proof test (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position would be to use the techniques developed for the Fort St. Vrain fuel production. The consequence of this action would be increased risk that the capability of the methods developed for FSV carbide fuel would not be adequate to show that the required high quality was achieved in oxide fuel production. It would therefore be necessary to design the plant to accept a lower quality level of fuel.

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Originator Date

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DATE: 3/11/87

UCO AND ThO_2 FUEL SCRAP AND WASTE HANDLING DEVELOPMENT
DDN M.07.22
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

In order to have a fully qualified fuel manufacturing process, techniques must be developed to recycle or dispose of fresh fuel scrap and waste products in an economical and safe manner. Since the processes for dealing with scrap and waste from each process unit operation will be similar, a single task has been designated to provide the needed technology.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high-quality fuel compacts for inclusion in prismatic fuel elements.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 7: Processes are available for manufacturing high-integrity coated fuel particles.

F1.1.4.1.1.2.1.1.1 "Retain Radionuclide in Fuel Kernels," Assumption 5: Processes are available for depositing-high integrity TRISO coatings on oxide-based fuel kernels.

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 7: Processes are available for depositing high-integrity TRISO coatings on oxide-based fuel kernels.

1.2 Current Data Base Summary

Substantial experience exists on the production of oxide-based fuels by the gel supported precipitated (GSP) and the sol gel processes at GA. TRISO fuel particle coating and fuel rod manufacturing experience is also extensive. Every one of those processes will produce substantial amounts of liquid and solid waste streams. To improve the economics of fuel manufacturing what is required is development of waste volume reduction and handling procedures. In order to provide adequate accountability of Special Nuclear Material (SNM), improved software and procedures that provide real time display of the distribution of SNM must be developed.

1.3 Data Needed

In order to define the overall costs for developing fuel for the MHTGR at pilot plant scale, it is necessary to define process flowsheets with the overall waste streams, SNM material accountability, and waste handling treatments. Waste treatment such as recycle or disposal will have substantial impact on fuel costs. Process specifications with the required instrumentation for waste monitoring and SNM material accountability are required for demonstration of the processes at pilot plant scale. All fuel process development must include waste handling, as well as SNM safeguards data on a system basis to develop scale-up unit for the overall fuel manufacturing cost estimates. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Production Module Throughput	$\geq$ 5 kg HM/day
Process Yield (kernels + coating + rods)	$\geq$ 75%
Waste Production Rate	2470 ℓ (650 gal) liquid/5 kg HM throughput
	420 ℓ (2 barrel) solid /5 kg HM throughput

2. DESIGNER'S ALTERNATIVES

Alternative to the above proposed approach are:

2.1 Utilize existing SNM control procedures and processes for waste and effluent handling which were developed for FSV fuel production.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The development effort will include the necessary data on waste handling, effluent control, and SNM material accountability to define fuel manufacturing costs at pilot plant scale. Process development for this task will include (a) development of processes for scrap treatment, waste handling, effluent control, and SNM safeguards for oxide-based fuel pilot plant processes; (b) build and demonstrate pilot plant units; (c) preparation of process and equipment specifications for these items with the necessary process flowsheets; and (d) preparation of fuel product cost estimates. This approach will provide specific process data and direct cost data for these oxide-based LEU fuels where only estimates based on work with HEU fuels now exist.

Alternative 2.1 was not utilized because FSV does not have large liquid side streams. Also, FSV safeguards do not provide the rapid displays of SNM inventory distribution that are now required.

4. SCHEDULE REQUIREMENTS

The processes development for handling wastes and effluent must be completed and documented one year before start of fuel production for the first core (9/91).

5. PRIORITY

Urgency: 1
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Existing FSV processes and systems would have to be used. Use of existing processes and systems would impose a severe economic penalty on fuel manufacturing. Use of an architect-engineering firm to redesign the existing facilities would result in a high risk schedule delay and significantly increased cost because of uncertainties in the details of the waste and effluent handling processes.

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ThO_2 FERTILE KERNEL PROCESS DEVELOPMENT
DDN M.07.23
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

In order to facilitate the nuclear design of the HTGR core with minimum power peaking coupled with favorable economics and safe shutdown margins, a Th containing particle is required. Prior work has shown that the low strength and density of current quality ThO_2 contributes to the TRISO particle SiC defect fraction in excess of 4 x 350 MW(t) plant requirements. Process and equipment development is required to produce ThO_2 with increased strength and density in order to meet 4 x 350 MW(t) plant requirements.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles,"
Assumption 7: Processes are available for manufacturing high-integrity coated fuel particles.

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels,"
Assumption 5: Processes are available for manufacturing oxide-based fuel kernels.

1.2 Current Data Base Summary

A full-scale thorium oxide (ThO_2) kernel production pilot line was operated for several years at GA in the late 1970s. This ThO_2 kernel line was replaced by other facilities in the early 1980s. Recent TRISO coating development studies with ThO_2 materials produced on the early GA ThO_2 kernel line with a low sintering temperature (1200°C) have shown that these materials had a low density and were mechanically too weak to remain intact during coating. The fragmentation of kernels led to a high defect fraction. Improving the mechanical and thermal properties of the ThO_2 kernels to achieve sufficient strength to survive coating and produce low defect coated particles will require additional ThO_2 kernel process development.

1.3 Data Needed

Development of a process to produce high strength ThO_2 kernels for the MHTGR is needed. Process and equipment specifications for the improved ThO_2 kernel pilot plant line are needed to document the work and guide future production. Thorium oxide kernels produced with the improved ThO_2 kernel line must be coated and fabricated into fuel rods for irradiation testing and validation of the successful development of the process. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

The ThO_2 kernel production process must demonstrate production of kernels with the following parameters:

Composition	ThO_2
Density	≥ 9.5 Mg/m^3
Diameter	≤ 500 μm
Throughput	≥ 40 kg/day
Yield	$\geq 90\%$

2. DESIGNER'S ALTERNATIVES

An alternative to the above described task is:

2.1 Delete Th from the fuel cycle and accept less desirable neutronic characteristics.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Development of ThO_2 was selected. The addition of the Th to the core imparts desirable neutronic characteristics which yield lower fuel temperature because of better fuel zoning and a lower positive reactivity effect during a water ingress event. The ThO_2 kernel pilot line will be assembled with the existing equipment from the prior ThO_2 line and existing UCO line to the extent possible. The major change will be a sintering furnace capable of 1800°C rather than the 1200°C limit in the prior process line. After the ThO_2 pilot line is operational, kernels will be manufactured using an existing sol-gel process and high-temperature sintering. The kernels will be subjected to TRISO coating processes and irradiation testing to ensure that low defect fraction fuel particles can be made which meet MHTGR performance requirements.

Alternative 2.1 was not selected because of adverse neutronic effects.

4. SCHEDULE REQUIREMENTS

A prototype process should be in place to provide fuel for the final irradiation-proof test prior to the submission of the PSSAR (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: L
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

If this task is not carried out, the fallback position would be to remove thorium from the fuel cycle. The consequence of removing Th from the fuel cycle would be increased fuel temperature or reduced outlet temperature and increased positive reactivity insertion during moisture ingress events.

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FUEL PROOF TEST
DDN M.07.24
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

In order to assure that the fuel specification, fuel process, and fuel design have been adequately defined and perform in accordance with licensing and design claims, a fuel proof test is needed.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high-quality fuel compacts for inclusion in prismatic fuel elements.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 7: Processes are available for manufacturing high-integrity coated fuel particles.

F1.1.4.1.1.2.1.1.1 "Retain Radionuclides in Fuel Kernels," Assumption 5: Processes are available for manufacturing oxide-based fuel kernels.

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings, "Assumption 7: Processes are available for depositing high-integrity TRISO coatings on oxide based fuel kernels.

F3.1.1.2.1 "Control Transport from Core," Assumption 7: Processes are available for manufacturing high-quality fuel compacts.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 5: Processes are available for manufacturing high-integrity coated fuel particles and compacts.

F3.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 4: Processes are available for depositing high-integrity TRISO coatings on oxide based fuel kernels.

1.2 Current Data Base Summary

Irradiation performance data are available from testing of TRISO HEU UC_2/ThO_2 and early TRISO LEU UCO/ThO_2. Fuel performance models are based on these data. The final demonstration line for fabrication of TRISO UCO/ThO_2 fuel has not yet been fully developed.

1.3 Data Needed

Data showing fuel failure fraction as inferred from Kr-85m and Xe-133 release and metallic release (Cs-137) are needed to confirm that fuel from the final demonstration line which meets the Fuel Product Specification and has been manufactured in accordance with the Fuel Process Specification exhibits mean observed failure at 95% confidence within 4X of that predicted by the fuel performance models at 50% confidence.

Since the fuel coatings provide a primary barrier to fission product release, proof data are needed to show that the optimized, final demonstration line processes deliver fuel with the required performance. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are required, as noted above, which will be applicable to the following range of service conditions:

Reference U.S. Fuel	TRISO UCO/ThO_2 Fuel Compacts, 12.5 mm diameter x 50 mm, in H-451 Graphite
Fuel Burnup Range	
o Fissile	20% to 22% FIMA
o Fertile	2% to 3% FIMA
Irradiation Exposure	4.0 to 5 x 10^{25} n/m^2 (E>29 fJ)
Fuel Temperatures	1000°C to 1250°C
Thermal Cycling	[TBD]
Coolant/Impurity Levels	Helium with $\leq$10 ppm Total Oxidants

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Utilize data from the initial demonstration line capsule test and correlate with quality of final demonstration line fuel.

2.2 Argue that the U.S. and FRG HEU and LEU fuel irradiation data base and process line experience, including that from the FSV HTGR, is directly applicable to the reference UCO/ThO_2 fuel system and no additional design qualification is needed.

3. SELECTED APPROACH AND EXPLANATION

Obtain irradiation performance data from fuel made on the fully developed demonstration line in accordance with the Fuel Product and Process Specifications. Alternative 2.1 leaves too much risk of nonconformance to fuel performance requirements. Alternative 2.2 does not provide a credible data base for the reference fuel system.

4. SCHEDULE REQUIREMENTS

The test must be completed and documented one year prior to the FSAR submittal (9/92).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to utilize data from the initial demonstration line capsule test based on fuel from an incompletely developed demonstration line which has been upgraded to the reference quality by special screening. The consequences are expected to be (1) difficulties and delays in licensing while technical arguments of the similarities between initial and final UCO/ThO_2 fuel are debated and (2) increased risk that the initial core will not perform as well as predicted because of unanticipated changes in fuel from the completed process line.

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Originator Date

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DEVELOPMENT OF PERFORMANCE MODELS FOR DEFECTIVE PARTICLES
DDN M.07.25
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Defective particle failure is a major contributor to circulating activity. The models used to predict failure of defective particles are needed to provide adequate confidence in core design predictions.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 5: Reference fuel failure models are sufficiently accurate to within a factor of [4] at 95% confidence.

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 5: Reference fuel failure models in fuel data base are sufficiently accurate to within factor of [4] at 95% confidence.

F2.1.4.1.1.2.1.1 "Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption 3: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

F2.1.4.1.1.2.1.1.2 "Protect the Capability to Retain Radionuclides with Particle Coatings," Assumption 3: Reference fuel failure models are sufficiently accurate to within a factor of [TBD] at 95% confidence.

F3.1.1.2.1 "Control Transport from Core," Assumption 6: Adequate data is available to predict fuel performance under transient conditions.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 3: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of 4x at 95% confidence.

F3.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 2: Reference fuel failure models are sufficiently accurate to within a factor of 4x at 95% confidence.

1.2 Current Data Base Summary

Indirect performance data exist for TRISO UCO/ThO_2 particles with variable levels of assorted defects. These indirect data and fundamental material properties have been used to derive analytical models which predict failure of particles with one or more defective layers. These particles are the primary source of failed fuel during irradiation. Overall core performance data from the FSV experience with HEU carbide fuel indicates that the defective particle failure models are overly conservative.

1.3 Data Needed

Data should provide defective particle failure models so that the observed mean failure at 95% confidence is within 4X of failure predicted at 50% confidence. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are required, as noted above, which will be applicable to the following range of service conditions:

Defects of Interest	o Missing buffers o Missing or defective SiC layers with intact OPyC layers o Heavy metal dispersion in the buffer layer
Fuel Burnup Range	
o Fissile	0% to 22% FIMA
o Fertile	0% to 3% FIMA
Irradiation Exposure	0 to 5 x 10^{25} n/m^2 (E>29 fJ)
Fuel Temperatures	700°C to 1250°C (Normal Conditions) 1200°C to 1800°C (Cooldown Conditions)
Coolant/Impurity Levels	Helium with $\leq$10 ppm Total Oxidants

2. DESIGNER'S ALTERNATIVES

An alternative to the acquisition of the above described data is:

2.1 Predict the performance of defective particles with present conservative models derived from fundamental materials properties and design for the predicted high failure rates of defective particles.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Acquire gaseous and metallic fission product release data from TRISO fuel with missing or defective layers and update the defective particle performance models to be consistent with the expanded data base. The data are expected to confirm the indications from FSV that the failure rates of defective particles are less than predicted with the existing models.

Alternative 2.1 was not selected because it would result in too much conservatism in performance prediction with costly fuel quality specification or reactor design needed to retain fission products.

4. SCHEDULE REQUIREMENTS

Data should be acquired and analyzed so that model modification can take place prior to start of the final design phase. (9/89)

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to rely on existing models for predicting the performance of defective fuel particles. The consequence is expected to be a higher failure fraction prediction than will actually be observed. Unnecessary conservatism in the performance models will result in unnecessarily stringent fuel product specifications or costly reactor design such as an attendant requirement for a containment building.

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Originator Date

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DATE: 3/11/87

VALIDATION OF FUEL PERFORMANCE MODELS UNDER NORMAL OPERATING CONDITIONS
DDN M.07.26
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The failure of reference particles in fuel rod, must be limited to extremely low fractional values in order to meet design requirements. The models used to predict failure of reference particles under normal irradiation conditions must be validated to provide the necessary confidence in core design.

1.1 Summary of Function/Title/Assumptions

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 5: Reference fuel failure models are accurate to within a factor of [4] at 95% confidence.

F1.1.4.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 5: Reference fuel failure models are accurate to within a factor of [4] at 95% confidence.

F3.1.1.2.1 "Control Transport from Core," Assumption 5: Validated methods and data are available to adequately assess fuel failure, fission product transport, and release from the nuclear steam supply system.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles, "Assumption 3: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of 4x at 95% confidence.

F3.1.1.2.1.1.2 "Retain Radionuclides with Particle Coatings," Assumption 2: Reference fuel failure models are sufficiently accurate to within a factor of 4x at 95% confidence.

1.2 Current Data Base Summary

The current irradiation performance models for reference fuel is based to a large extent on previous experience with HEU carbide fuel. In terms of direct measurements of performance of the behavior of reference TRISO LEU UCO/ThO_2 particles, limited data from scoping tests exist which show a range of about 5X between observed and predicted gas release at peak exposure. Because the existing data base represents fuel with about 10X greater defects and contamination than allowed in 4 x 350 MW(t) HTGR fuel, these results do not satisfy the requirements for validation of the 4 x 350 MW(t) HTGR thermal and pressure vessel fuel performance models.

1.3 Data Needed

Data showing fuel failure fraction, as inferred from Kr-85m and Xe-133 release and metallic release (Cs-137), are needed to provide validation of performance models and support licensing of the HTGR. These data are needed to validate that the observed mean failure at 95% confidence is within 4X of the 50% confidence mean failure predicted by thermal and pressure vessel fuel performance models which are used in core design. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are required, as noted above, which will be applicable under the following range of service conditions:

Fuel Burnup Range	
o Fissile	0% to 22% FIMA
o Fertile	0% to 3% FIMA
Irradiation Exposure	0 to 5 x 10^{25} n/m^2 (E>29 fJ)
Fuel Temperatures	700°C to 1250°C
Thermal Cycling (Normal Operation)	[TBD]
Coolant/Impurity Levels	Helium with <10 ppm Total Oxidants

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Argue that models that are not validated for reference LEU UCO/ThO_2 fuel and that are based on limited, accelerated testing conditions and on HEU carbide fuel are applicable to the 4 x 350 MW(t) Prismatic HTGR reference fuel.

2.2 Utilize FRG fuel performance data and argue that these data are directly applicable to U.S. prismatic fuel without validation.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Fission product release data from fuel made on an incompletely developed demonstration line and representative of real time irradiation conditions will be obtained. The demonstration line may be incomplete, but developed to the point that representative 4 x 350 MW(t) HTGR quality fuel can be obtained by screening or upgrading, as required. These data will validate the normal condition thermal and pressure vessel fuel performance models for the reference fuel.

Alternative 2.1 was not chosen because of the high risk that validation will be required and licensing delay could result from failure to get early validation. Alternative 2.2 was not chosen for the same reasons given for Alternative 2.1. In addition the FRG fuel performance data would suffer additional risk of not complying with 10CFR50 requirements.

4. SCHEDULE REQUIREMENTS

The data for validation of normal condition performance models must be obtained prior to start of the final design phase (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position would be to argue that the data base on the reference UCO/ThO_2 fuel system is not needed due to the existing HEU carbide data base. The designer would rely on existing models for predicting the normal condition performance of reference fuel. The consequences are expected to consist of (1) difficulties and delays in licensing while technical arguments of the similarities of HEU carbide and UCO/ThO_2 fuel are debated and (2) increased risk that the performance of the initial core will not be as good as predicted because the models under predict failure.

D.W. Stansfield

Originator Date

R F Turner 3/17/87

Department Manager Date

[illegible] for G.C. Bramblett 3.25.87

Manager, Project Operation Date

DATE: 3/11/87

VALIDATION OF FUEL PERFORMANCE MODELS UNDER CORE CONDUCTION COOLDOWN CONDITIONS
DDN M.07.27
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The performance of fuel particles during a conduction cooldown thermal transient with and without moisture ingress must be known with good precision in order to assure that passive safety goals have been achieved in the design. The models used to predict particle failure during an accident must be validated in order to provide the necessary confidence in design.

1.1 Summary of Function/Title/Assumptions

F2.1.4.1.1.2.1.1 "Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption 3: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

F2.1.4.1.1.2.1.1.2 "Protect Capability to Retain Radionuclides with Particle Coatings," Assumption 3: Reference fuel failure models are sufficiently accurate to within a factor of [TBD] at 95% confidence.

F3.1.1.2.1 "Control Transport from Core," Assumption 6: Adequate data is available to predict fuel performance under transient conditions.

F3.1.1.2.1.1, "Retain Radionuclides in Fuel Particles," Assumption 3: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within factor of 4x at 95% confidence.

F3.1.1.2.1.1.2, "Retain Radionuclides with Particle Coatings," Assumption 2: Reference fuel failure models are sufficiently accurate to within a factor of 4x at 95% confidence.

1.2 Current Data Base Summary

Prior data taken in support of fuel performance models to predict accident behavior were primarily derived under dry conditions from unbonded particles rather than compacts, relatively small sample sizes, and a variety of fuel types. There is a lack of data in the 1200°C to 1800°C temperature range because most measurements were made at higher temperatures characteristic of large HTGR core heatup accidents. First-order models have been developed with these data, including a joint U.S./FRG accident condition model based on both U.S. and FRG data. A complete uncertainty analysis for this model has not been conducted, but the uncertainty in the model has been estimated at 12X at the 95% confidence level for the FRG fuel in the range of conditions of interest.

1.3 Data Needed

Data are needed to describe failure of coated particle coatings as indicated by the release of Cs-137 from reference fuel rod compacts under transient conditions characteristic of pressurized and depressurized conduction cooldown events in the 4 x 350 MW(t) MHTGR. While it is known that fuel rod matrix and OPyC coatings will oxidize under air and moisture ingress conditions, a validation of the models used to predict the rate of oxidation is needed. Furthermore, the oxidation rate and subsequent failure of the SiC under high temperature high moisture conditions is not known for irradiated fuel particles, so data is needed to make predictions on TRISO particle failure under high moisture conditions.

The data must validate that the physical models are (a) suitable for use in design codes covering the range of expected service conditions, (b) capable of predicting the release of the key radionuclides noted above from the reference fuel so that the observed release at 95% confidence is within a factor of 10X or 4X of the predicted mean at 50% confidence for fission product metals and gases, respectively. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are required as noted above, which will be applicable to the following range of service conditions:

Fuel Burnup Range	
o Fissile	20% to 22% FIMA
o Fertile	2% to 3% FIMA
Irradiation Exposure	4 to 5 x 10^{25} n/m^2 (E > 29 fJ)
Environment	He; He/CO/H_2; CO/N_2
Fuel Temperatures/Time	800° to 1000°C (Depressurized)/1000 h (dry) 1600° to 1800°C (Depressurized)/100 h (dry) 800° to 1250°C (Pressurized)/100 h (wet)

Reference Transients	Pressurized conduction cooldown with and without water ingress, Depressurized Conduction Cooldown (Ref. PSID Chap. 15 Doc. No. HTGR-86-024)
Pressure	0.101 MPa (1 atm)
Linear Flow Rate	[TBD] to [TBD] cm/s
Coolant/Impurity Levels	2 to 10^4 ppm H_2O Total oxidants $<10^4$ ppm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Use existing performance models based on data developed primarily for the large HTGR plus the limited data currently available in the 1200°C to 1800°C temperature range for loose particles. Accept the large uncertainty resulting from extrapolations and the lack of data from reference fuel in the temperature, pressure, and environmental ranges of interest.

2.2 Utilize FRG performance data to expand the data base and argue that these data are directly applicable and validate reference U.S. prismatic fuel.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Fission product release data from irradiated fuel elements will be obtained as a function of time and temperatures simulating Conduction Cooldown conditions. These data will allow the design models describing performance of fuel under conduction cooldown conditions to be validated and the uncertainty interval reduced to the required precision. Data from fuel compacts will help characterize any fission product retention effects from matrix sorption and matrix-particle interaction which is not included in existing data from unbonded particle heating.

Alternative 2.1 was not chosen because it leaves too much uncertainty in the models. Also there may be licensing delay because of a firm requirement for validation of the models with reference fuel and conditions. Alternative 2.2 was not chosen because of the high risk that application of FRG data to U.S. fuel would not be accepted without at least a modest amount of data derived from U.S. fuel under the conditions of interest.

4. SCHEDULE REQUIREMENTS

Complete characterization of early pilot line fuel at beginning of final design (10/89). Validated design models reflecting the expanded data base are needed six months before the start of FSSAR documentation (9/91) (controlling schedule requirement).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

If this work is not performed, it would be necessary to design with very large uncertainties in fuel performance models so that specification of an unnecessarily conservative fuel and core design may result. Thus the fallback position is use of existing fuel performance data with the consequences consisting of (a) a design with unnecessary conservatism to account for uncertainty, and (b) a worst case consisting of a containment building to assure acceptable fission product retention.

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Originator Date

R F Turner 3/17/87
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DATE: 3/11/87

CHARACTERIZE FUEL COMPACT DIMENSIONAL CHANGE PROPERTIES
DDN M.07.28
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Design assumptions of no deleterious interaction of fuel compact and fuel block require a fuel compact dimensional change model with improved accuracy. More accurate temperature predictions also depend upon knowledge of fuel compact dimensional change (gap width). Therefore, an improved fuel compact dimensional change model is needed.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.1 "Transfer Heat from Fuel to Heat Transfer Surface," Assumption 1: Thermal properties of fuel compacts are known and adequate. Assumption 4: Thermal properties data base for fuel compacts is adequate.

F1.1.2.1.2.2 "Maintain Core Coolant Passages Geometry," Assumption 1: Thermal property data base is adequate.

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity," Assumption 7: The existing fuel compact thermal/mechanical property data base is adequate.

1.2 Current Data Base Summary

The current fuel compact dimensional change model was developed in 1979 (904308/A) and does not contain the fuel compact dimensional change data generated in the more recent capsules containing TRISO/TRISO fuel particles; HRB-14, HRB-15A, HRB-16, HRB-17, HRB-18, and R2-K13. It has been shown that the current dimensional change model does not always well predict the actual dimensional change of the recent test capsule fuel compacts. The difference between measured and predicted compact diameters after irradiation are as large as 1.4% of the compact diameter. The minimum fuel hole/fuel compact gap during service is only 2% of the fuel hole diameter so an improved fuel compact dimensional change model is needed to reduce the difference between observed and predicted diameter to $\leq 1\%$ of compact diameter. This will result in high confidence that no compact-graphite interference will take place.

Fuel compact dimensional change data from capsules HRB-14, HRB-15A, HRB-16, HRB-17, HRB-18, and R2-K13 exists for containing TRISO/TRISO fuel particles for temperatures ranging from 700° to 1200°C, fast fluence exposures from 3.3 to 7.8 x 10^{25} n/m^2 (E > 29 fJ) HTGR, and for fuel compact shim contents ranging from 23 to 40 volume percent. The dimensional change data from these capsules needs to be analyzed and a revised fuel compact dimensional change model developed on the basis of this and additional data to be obtained from irradiation capsules HRB-19, HRB-20, and HRB-21.

1.3 Data Needed

An improved fuel compact dimensional change model is needed because the lack of agreement between the measured and predicted fuel compact dimensional change measurements from recent irradiation capsules exceeds 1% of the rod diameter.

Sufficient irradiation data on fuel rod dimensional change already exist to permit a preliminary model to be developed using data from irradiation capsules HRB-14, HRB-15A, HRB-16, HRB-17, HRB-18, and R2-K13. A final fuel rod dimensional change model would be completed from data generated from capsules HRB-19, -20, and -21.

Quality assurance must be in accordance with requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Data are required, as noted above, which will be applicable to the following range of service conditions:

Reference U.S. Fuel	TRISO UCO/ThO_2 Fuel Compacts, 12.5 mm diameter x 50 mm, in H-451 Graphite
Fuel Burnup Range	
- Fissile	20% to 22% FIMA
- Fertile	2% to 3% FIMA
Irradiation Exposure	4.0 to 5 x 10^{25} n/m^2 (E > 29 fJ)
Fuel Temperatures	1000° to 1250°C
Thermal Cycling	[TBD]
Coolant/Impurity Levels	Helium with $\leq$10 ppm Total Oxidants
Accuracy of Fuel Compact OD and Length Measurement Before and After Irradiation	$\leq$0.025 mm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisitions of the above described data are:

2.1 Use the existing fuel compact dimensional change models which are based primarily on a TRISO (fissile)/BISO (fertile) fuel system and result in large uncertainties. This will result in increased risk of excessive compact-block interaction and licensing delays if compact-block interaction becomes an issue.

2.2 Modify the model after analysis of the existing TRISO/TRISO data base (capsules HRB-14, 15A, 16, 17, 18, and R2-K13) but do not incorporate the results from irradiation capsules HRB-19, -20 and -21 which are to be irradiated in the near future.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to use the existing data (capsules HRB-14, 15A, 16, 17, 18 and R2-K13) to modify and upate the existing fuel compact dimensional change model. After analysis of compact dimensional change results from future capsules HRB-19, 20, and 21 the fuel compact dimensional change model will again be updated to provide the model used in final core design.

Alternative 2.1 and 2.2 were not chosen because the resulting uncertainty in dimensional change would be too great.

4. SCHEDULE REQUIREMENT

The preliminary model revision should be completed six months prior to the start of the final design phase (3/89); the final model update is needed one year prior to the FSSAR (9/92) (controlling schedule requirement).

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty of existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Use the existing model in core design is the fallback position. This will result in (1) increased risk of unsatisfactory fuel compact-fuel block interaction, (2) increased risk of higher than predicted fuel temperatures because of larger than predicted compact-graphite gap, and (3) increased risk of licensing delay if compact-block interaction becomes an issue.

[74-38]

OMA 3/17/87

W. J. Scheffel 3/17/87

Originator Date

R F Turner 3/17/87

Department Manager Date

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G. R. Bramblett 3.25.87

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DATE: 2/27/87

DETERMINE CONDUCTION COOLDOWN TO RCCS
DDN M.08.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 57

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Conduction cooldown to the safety-related Reactor Cavity Cooling System (RCCS) provides a backup method for removing decay heat from the core if both main loop and shutdown cooling system (SCS) cooling fails. The decay heat is removed by natural convection, conduction, and radiative heat transfer to the RCCS. Unless effective heat transfer to the RCCS and heat removal takes place, allowable reactor core, reactor internal component, and reactor vessel temperatures cannot be met.

1.1 Summary of Function/Title/Assumptions

F2.0, "Maintain Plant Protection," Assumption 4. Data are available to adequately assess forced outages and investment risk.

F2.1.2, "Protect the Capability to Maintain Energy Transfer," Assumption 6. Methods will be available for the timely prediction of NSSS component behavior during loss of main and SCS cooling loops.

F2.1.4.1.1.2.1.1.2.3, "Maintain Alternate Cooling," Assumption 1. Validated methods and data are available for the prediction of fuel, core reactor internals, and reactor vessel temperatures.

1.2 Current Data Base Summary

The heat transport from the core to the RCCS for pressurized and depressurized conduction cooldown is calculated with computer codes PANTHER and TAC2D, respectively. Both codes are verified. A predecessor of PANTHER is used extensively in the aerospace industry, and TAC2D has been tested against a series of benchmark problems as well as numerical solutions published in the literature. Validation of heat transport modeling and calculations of component temperatures have not been performed. No experimental data on the heat transport process from the core to the RCCS exists.

1.3 Data Needed

The following data are needed to validate the analytical models:

a. Effective radial and axial core block thermal conductance and the effective core block heat capacity.

b. Detailed natural convection, conduction, and radiative heat transfer rates in the reactor vessel including natural convection flow in the reactor core and mixing in the upper core plenum.

c. Natural convection flow rates into the hot duct and steam generator vessel.

d. Natural convection and radiative heat transfer rates to the RCCS.

e. Flow rates inside the RCCS.

f. Temperatures in the following components: reactor core, control rods, core barrel, core support plate, reactor vessel including upper head, mid-section and lower head, cross duct, and RCCS.

These data should provide transient thermal models that predict component temperatures (°F) within +10% -0% error.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

The data are required to validate the conduction cooldown methods under the following service conditions:

Reactor configuration	o Annular reactor core with prismatic graphite fuel blocks and reflector blocks, control rods, core barrel, core support plate, reactor vessel and cross duct, and RCCS.
Reactor coolant	o Helium
Reactor conditions	o Conduction cooldown; Pressurized Depressurized
Pressure	o 1150 psi to atmospheric
Fuel temperature	o 3200°F maximum 2000°F average
Fuel block irradiation exposure	o Up to 4.5×10^{25} n/m^2 (E > 29 fJ)
Irradiation temperature	o Up to 2200°F
Plant elevation	o [100 ft] above mean sea level

Outside air temperature	o	−45°F to 110°F dry bulb
	o	82°F (max) wet bulb
RCCS conditions	o	Natural convection
Conduction cooldown time period	o	Up to ~1000 h

2. DESIGNER'S ALTERNATIVES

The following alternative is considered:

2.1 Predict the thermal data with present methods derived from fundamental materials properties and accept the uncertainties in the predictions. Design affected components conservatively by selecting more temperature resistant materials and/or reduce the module power output, thereby reducing the probability of component damage.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Obtain the data needs at service conditions from either (1) an integrated test on a simulated nonnuclear scale model of the 350-MW(t) reactor and RCCS, or (2) separate tests on component models. Either approach can provide realistic heat transfer parameters associated with the conduction cooldown event. The integrated test approach, however, will provide the more comprehensive method for code validation. Compared to the alternative design approach in Section 2.1, code validation with test data can remove excess conservatism in the design since the uncertainties in the analysis are reduced.

4. SCHEDULE REQUIREMENTS

These data are required before the end of the preliminary design phase to aid final design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NON-EXECUTION

If the selected task is not performed, the fallback position is alternative 2.1 which relies on existing analytical models and conservative design measures to assure adequate design margins. This could require more temperature resistant materials for the reactor vessel, control rods, and reactor internal components or result in a reduced core output. Consequently, power costs would increase.

Bernt Olsam 3/19/87

Originator Date

Donald A. Blaily 032487

Department Manager Date

for

G.R. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

VALIDATION OF DESIGN METHODS FOR GRAPHITE CORROSION
DDN M.10.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes used to predict the extent of corrosion of graphite components by coolant impurities must be validated to have the specified predictive accuracies for normal operating conditions and for moisture ingress events.

Associated data needs: DDNs M.10.18.08 and M.10.18.09.

1.1 Summary of Function Number/Title/Assumptions

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity," Assumption 5: The existing design methods and computer codes for calculating graphite corrosion are accurate within a factor of 3 at 95% confidence.

F3.1.1.2.1.1.2.1.1.3 "Maintain Controllable Geometry," Assumption: The existing design methods and computer codes for calculating graphite corrosion are accurate within a factor of 3 at 95% confidence.

1.2 Current Data Base Summary

Calculational methods have been developed to predict graphite corrosion in a HTGR environment. These methods include the computer codes OXIDE and HYDROBURN. The codes are based on the current data base on corrosion of various graphites developed in a laboratory environment. The validity of the models for graphite corrosion in the HTGR environment have not been thoroughly assessed although limited comparisons with surveillance data from FSV have been made with apparent good success.

1.3 Data Needed

Validation of the integrated models and computer codes used to predict graphite corrosion in the HTGR core under normal operation and during steam and air ingress events are needed in order to assure that the predictive methods are accurate to within 3x at 95% confidence. Particular attention must be given to transport of coolant impurities in fuel element graphite and to the effect of catalysis by graphite impurities and fission metals. The data base used for code validation must be independent from the data from which the individual correlations in the overall design method (effective diffusivities, reaction kinetics, etc.) were originally derived, in accordance with software standard IEEE Standard 730-1984 and software definitions in NUREG-0856. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions are given below:

a. Normal Operation

Environment	Helium
Maximum fast fluence (E > 29 fJ HTGR)	[5 x 10^{21}] n/cm^2
Maximum gamma flux	[TBD] MeV/cm^2-s
Primary coolant temperature range	[120 - 700] °C
Fuel element temperature range	[120 - 950] °C
Reflector element temperature range	[120 - 900] °C
Maximum time averaged coolant impurity levels	[2] ppm H_2O [5] ppm CO [2] ppm CO_2 Total Oxidants < [10] ppm maximum but not to exceed [600] ppm days per year
Helium coolant pressure	1-63 atm

b. Moisture Ingress Conditions

	Maximum Concentration (ppmv)
Moisture ingress with steam generator dump failure (DBE-9)	660
Moisture ingress with moisture monitor failure (DBE-8)	18,000

	Amount of Water Leaking into Reactor Vessel
Moisture ingress without steam generator dump (SRDC-6, 7)	1820 lb

c. Air Ingress Condition

	Amount of Air Ingress
Depressurized conduction cooldown (SRDC-10)	21 lb-mole

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Complete the design on the basis that the methods are acceptable without validation.

2.2 Eliminate the need to validate the design methods by including sufficient margin in the design to account for the uncertainties.

2.3 Impose tighter tech specs on primary coolant oxidant levels.

2.4 Use a higher purity, more corrosion-resistant graphite.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain a data base on the corrosion of graphite components in support of code validation under conditions expected in a modular HTGR. Alternative 2.1 would involve the risk of rejection in licensing. Alternative 2.2 may require excessively large margins in the design to account for uncertainties in the design methods. Alternative 2.3 would have an adverse effect on plant availability. Alternative 2.4 would lead to large increases in the development costs.

4. SCHEDULE REQUIREMENTS

Preliminary data by [3/89], six months prior to PSSAR submittal and final data by [9/92], one year prior to FSSAR submittal.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

A combination of Alternatives 2.1, 2.2, and 2.3, with the necessity of added conservatism in the design to compensate for calculational uncertainties. A weakened licensing position will result from this uncertainty. Another consequence of nonexecution will likely be unnecessarily restrictive tech specs on primary coolant impurities with an attendant adverse impact on plant availability.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CONTROL ROD DRIVE (CRD) DESIGN VERIFICATION
DDN M.10.12.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The control rod drive assembly must be highly reliable and perform repeatedly the required functions during its design life. Qualification testing of the neutron control assembly is described in DDN M.10.12.08.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods," Assumption 4: Control rod drives can be controlled with a repeatable positional accuracy that is sufficient for reactor control and with a reliability of control rod drive operation of [TBD].

F2.1.1.2.4.3 "Execute Commands," Assumption 1: Control rod drives respond correctly to shutdown command signals with reliability of [TBD] and have a probability of spurious scram $\leq$[TBD].

1.2 Current Data Base Summary

The reference control rod drive design is significantly different from that of FSV because the rod is heavier, the stroke is longer, there is no orifice assembly, the entire assembly is within the primary system, and because of improvements due to FSV experience. Data on the proposed DC torque motor were gathered in experiments for the large HTGR, but no data on the other features are available.

1.3 Data Needed

Design features requiring validation testing are reliability, speed of rod motion under normal and scram conditions, accuracy of rod positioning and strength of the assembly.

Data on the reliability and performance of the CRD under long-term exposure to the reactor environment are needed. Experimental data acquired should establish for the prototype design (1) operating speed versus current load, (2) scram speed versus load resistor size, (3) change in internal friction with age, (4) positioning accuracy, (5) nonlinearity of drum wrap-up, (6) instrumentation and control operating characteristics, and (7) response time. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Radiation Level	[TBD]
Helium Pressure	924 psia
Impurity Level	<[10] ppmv oxidants
Temperature	[150-200]°F
Control Rod Weight	206 lb

2. DESIGNER'S ALTERNATIVE

The following alternative is available:

2.1 Rely on analysis and test CRD during neutron control assembly qualification test.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Performance data on a full-scale control rod drive assembly will be obtained for all expected conditions using simulated control rod channels and operating environment. The selected approach allows deficiences in the reference design to be corrected in sufficient time to avoid expensive production line retooling. The full-scale test is reasonably achievable cost-wise and provides a high level of confidence in the applicability of the experimental data. Alternative 2.1 carries high risk of delays and expensive design changes if performance or reliability of the CRD were not confirmed until the qualification testing which occurs late in the design schedule.

4. SCHEDULE REQUIREMENTS

Completion 6 months prior to finish of plant final design phase (3/93).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to validate performance and reliability of the CRD during the neutron control assembly qualification test. This would increase risk of schedule delays and/or cost impact if design deficiencies were identified during the neutron control assembly qualification testing and major portions of the test had to be repeated.

Edwin P. Harvey 3/25/87
Originator Date

K J Turner 3/25/87
Department Manager Date

[illegible] for G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

RESERVE SHUTDOWN CONTROL EQUIPMENT (RSCE) DESIGN VERIFICATION
DDN M.10.12.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The RSCE must perform its functions in a highly reliable and timely manner. Qualification testing of the neutron control assembly is described in DDN M.10.12.08.

1.1 Summary of Function/Title/Assumptions

F3.1.1.2.1.1.2.1.1.2.3 "Execute Commands," Assumption 1: Reserve shutdown control equipment (each hopper) responds correctly to shutdown signals with reliability of [TBD] and has a probability of spurious operation ≤[TBD].

1.2 Current Data Base Summary

The proposed design of the RSCE is different from the one tested for and used at Fort St. Vrain because the FSV rupture disk has been replaced by fuse links and the hopper is larger. Testing data on the fuse link design showed it could be used in place of the pneumatic rupture disk, thereby justifying the design change.

1.3 Data Needed

Operational performance data on the release mechanism, controls, pellet flow, and channel configuration are needed to establish (1) response time, (2) material flow rates, (3) system reliability, and (4) power requirement for fuse link operation and periodic testing. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Radiation Level	[TBD]
Pressure	924 psia
Helium Impurity	<[10] ppm oxidants
Temperature	[150-200]°F

2. DESIGNER'S ALTERNATIVE

The following alternative is available:

2.1 Rely on verification of RSCE mechanism as part of the complete neutron control assembly qualification test.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Performance data on the prototype reserve shutdown control equipment will be obtained for all expected conditions and operating environment. The selected approach allows deficiencies in the reference design to be corrected in sufficient time to avoid expensive production line retooling. A full-scale test is reasonably achievable cost-wise and provides a high level of confidence in the applicability of the experimental data. Alternative 2.1 carries high risk of delays and expensive design changes if performance or reliability of the RSCE is not confirmed until the qualification testing which occurs late in the schedule.

4. SCHEDULE REQUIREMENTS

Completion 6 months prior to finish of plant final design phase (3/93).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NON-XECUTION

The fallback position is to validate performance and reliability during neutron control assembly qualification test. This would increase risk of schedule delay if design deficiencies were identified during the neutron control assembly qualification testing and major portions of the test had to be repeated.

Edwin C. Harvey 3/25/87
Originator Date

R. Turner 3/25/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

PRELIMINARY QUALIFICATION OF ELECTROMECHANICAL COMPONENTS OF THE NEUTRON CONTROL ASSEMBLY
DDN M.10.12.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

All electromechanical components of the neutron control assembly must be highly reliable and perform repeatedly their required functions during their design lives. Final qualification testing is described in DDN M.10.12.07.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods," Assumption 4: Control rod drives can be controlled with a repeatable positional accuracy that is sufficient for control and with a reliability of [TBD].

F2.1.1.2.4.3 "Execute Commands," Assumption 1: Control rod drives respond correctly to shutdown command signals with a reliability of [TBD] and have a probability of spurious scram that is $\leq$[TBD].

1.2 Current Data Base Summary

The design of the neutron control assembly is based upon experience with designs developed for FSV and various large HTGR plants over the past 20 years. However, because performance requirements such as stroke and power are significantly different for this design, data from the older designs are not applicable.

1.3 Data Needed

Reliability and performance data such as seismic response, operating speeds, effect of thermal aging, vibration and wear, accuracy and strength are needed under simulated operating conditions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal Atmosphere	Air/helium
Temperature	
Components located above gamma shield	[200°F]
Components located within refueling plenum	497°F
Pressure	924 psia

Radiation	[TBD]
Maximum horizontal acceleration (SSE)	[1.2 g]
Maximum vertical acceleration (SSE)	[1.0 g]

2. DESIGNER'S ALTERNATIVES

2.1 Rely on final qualification testing.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Full-scale test articles will be constructed and tested. The tests will be based on guidance from regulations and industry standards. The components selected will be those which are essential for operation of the assembly.

A full-scale test under expected conditions was chosen to determine deficiencies in the proposed design in sufficient time to make corrections prior to final qualification tests. This testing avoids expensive changes should the proposed design be inadequate.

Alternative 2.1 was not chosen because of the risk associated with basing the plant licensing schedule on the assumption that the equipment would pass the final qualification tests.

4. SCHEDULE REQUIREMENTS

Tests to be completed 12 months before the finish of the plant final design phase (9/92).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is Alternative 2.1. Nonexecution of the selected approach would introduce high risk to the licensing schedule due to reliance on the assumption that the final qualification tests would be satisfactory.

Edwin C. Harvey 3/25/87
Originator Date

R J Turner 3/25/87
Department Manager Date

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G.C. Bramblett 3.25.87
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DATE: 2/27/87

DEMONSTRATION OF REMOTE HANDLING AND MAINTENANCE FEATURES
OF THE NEUTRON CONTROL ASSEMBLY
DDN M.10.12.04
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Contamination and activation require that the neutron control assembly have features which permit remote handling and maintenance. These features must be tested to demonstrate adequacy. Qualification testing of the neutron control assembly is described in DDNs M.10.12.07, and M.10.12.08.

1.1 Summary of Function/Title/Assumptions

F.1.1.9.3.1 "Perform Neutron Control Assembly Maintenance," Assumption 1: Equipment must satisfy the reliability function.

1.2 Current Data Base Summary

The proposed Neutron Control Assembly is significantly different from that of Fort St. Vrain. However, the conceptual design of the remote handling features was based on the experience gained at the Fort St. Vrain Hot Service Facility. Nevertheless, data on the handling features of the new control assembly design are not sufficiently complete.

1.3 Data Needed

Data on various features of the assembly and the procedures and tools used for its handling are needed. Operations for the removal, alignment, and installation in the reactor vessel and reactor service facilities must be performed and evaluated. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal Atmosphere	Air
Pressure	14.7 psia
Temperature	120°F
Radiation	[TBD]

2. DESIGNER'S ALTERNATIVES

2.1 Demonstrate remote handling capability on a prototype assembly.

2.2 Verify remote handling and maintenance at the plant site.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Acceptability of the remote handling features of the control assembly will be verified by a series of tests in which the anticipated maintenance manipulations are performed on full-scale test components. The tests will be conducted on simulated reactor penetrations and in mock-ups of the service facility. Neither of the alternatives considered was selected because each introduces significant risk of plant licensing delays should either require modifications during late-in-schedule testing.

4. SCHEDULE REQUIREMENTS

Testing to be completed six months before the finish of plant final design phase (3/93).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative 2.1 is the leading fallback position if a prototype will be available on a timely basis. Nonexecution of the selected approach would increase the risk of having to make changes to the production units, thereby potentially causing delays in the licensing schedule, or requiring the development of new tools and procedures late in the plant schedule.

Edwin C. Harvey 3/25/87
Originator Date

R F Turner 3/25/87
Department Manager Date

Coulisman for
G.C. Bramblett 3.25.87
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DATE: 2/27/87

NEUTRON CONTROL ASSEMBLY FLOW AND LEAK DESIGN VERIFICATION
DDN M.10.12.05
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Helium flow up through the neutron control assembly must not result in overheating of drive system components, and flow through the neutron control assembly must be held to acceptable levels during maintenance operations.

1.1 Summary of Functions/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods" Assumption 1: Control assembly does not allow helium to leak into upper region of penetration to the extent that causes overheating.

1.2 Current Data Base Summary

As a result of high temperatures in the upper head of the FSV PCRV due to leaks through the control assemblies, a series of tests was run to determine flow paths and resistances through the control assembly. The data gathered in these tests are not applicable to the current design which has no orifice and is intentionally more leaktight. However, the methods used for the FSV tests are applicable to this DDN.

1.3 Data Needed

Data must be obtained on the flow resistances of the various flow paths under various combinations of misalignments and pressure differentials. These data will be used in analysis to predict convective heating and contamination of the upper portion of the refueling penetration and gas flow through the penetration. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Ambient Temp. of Control Rod Drive (°F)	150
Core Inlet Pressure (psia)	924
Core Inlet Temperature (°F)	497
Helium Flow Rate (lb/h)	1,248,100

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Rely upon analysis until hot flow tests are performed during commissioning.

2.2 Add limited helium flow test capability to neutron control assembly prototype qualification test.

2.3 Small scale tests with data extrapolation to full size unit.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A full-scale test rig will be instrumented to measure flow rates and pressure differentials. If necessary, results will be extrapolated to reactor conditions. Alternative 2.2 could cause substantial delay to the qualification test program. Alternative 2.1 could require substantial rework of all neutron control assemblies if design deficiencies are identified. Relying on small scale tests (Alternative 2.3) would increase the uncertainty in the test results.

4. SCHEDULE REQUIREMENTS

Data from this test are required six months prior to finish of plant final design phase (3/93).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The preferred fallback position is Alternative 2.3. Nonexecution could result in increased risk of control rod drive mechanism overheating, excessive dose rates to personnel during maintenance operations, or schedule delays if problems were detected during hot flow testing.

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Originator Date

R J Turner 3/25/87
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G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

ELECTROMECHANICAL COMPONENTS QUALIFICATION FOR THE NEUTRON CONTROL ASSEMBLY
DDN M.10.12.07
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

All electromechanical components of the neutron control assembly must be highly reliable and perform repeatedly their required functions during their design lives.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods," Assumption 4: Control rod drives can be controlled with a repeatable positional accuracy that is sufficient for control and with a reliability of [TBD].

F2.1.1.2.4.3 "Execute Commands," Assumption 1: Control rod drives must respond correctly to shutdown command signals with a reliability of [TBD] and have a probability of spurious scram that is [TBD].

1.2 Current Data Base Summary

The design of the neutron control assembly is based upon experience with designs developed for FSV and various large HTGR plants over the past 20 years. However, because performance requirements such as stroke and power are significantly different for this design, data from the older designs are not applicable.

1.3 Data Needed

Reliability and performance data such as seismic response, operating speeds, effects of thermal aging, vibration and wear, accuracy and strength are needed under simulated operating conditions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal Atmosphere	Air/helium
Temperature	
Components located above gamma shield	200°F
Components located within refueling plenum	497°F

Pressure	924 psia
Radiation	[TBD]
Maximum horizontal acceleration (SSE)	[1.2 g]
Maximum vertical acceleration (SSE)	[1.0 g]

2. DESIGNER'S ALTERNATIVES

2.1 Qualify design on the basis that sufficient prior data/experience exists to obtain license without need for qualification testing.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Components as-built on the production tooling will be tested. The tests will be in accordance with guidance from applicable codes, standards, and regulations. Alternative 2.1 was rejected because there is little chance of its success.

4. SCHEDULE REQUIREMENTS

Qualification program to be completed by the completion of plant final design phase (9/93).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is Alternative 2.1. This approach is almost certain to lead to licensing delays.

Edwin C. Harvey 3/25/87
Originator Date

R J Turner 3/25/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

NEUTRON CONTROL ASSEMBLY SEISMIC QUALIFICATION
DDN M.10.12.08
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The complete neutron control assembly including the reserve shutdown control equipment, control rod and control rod channel must maintain the ability to insert control material in a timely manner during and following a safe shutdown seismic event. In addition, there must be no degradation of any performance characteristic during or following an operating basis seismic event.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods," Assumption 4: Control rod drives can be controlled with a repeatable positional accuracy that is sufficient for reactor control and with a reliability of control rod drive operation of [TBD].

F2.1.1.2.4.3 "Execute Commands," Assumption 1: Control rod drives respond correctly to shutdown command signals with reliability of [TBD] and have a probability of spurious scram <[TBD].

F3.1.1.2.1.1.2.1.1.2.3 "Executive Commands," Assumption 1: Reserve shutdown equipment responds correctly (each hopper) to shutdown signals with reliability of [TBD] and has a probability of spurious operation (each hopper) that is ≤[TBD].

1.2 Current Data Base Summary

Configuration and components of the neutron control assembly are significantly different from those used in Fort St. Vrain. Also, data from tests of equipment for the large HTGR did not cover seismic events. Therefore, no seismic qualification data showing that the control assembly complies with IEEE-323 are available.

1.3 Data Needed

Data are needed to demonstrate the operability of the neutron control assembly during and following a seismic event in compliance with requirements specified in IEEE-323. Data acquired should include the control system response time, control material insertion time, and reliability. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Helium Pressure	924 psia
Impurity	<[10] ppm oxidants
Temperature	[150-200]°F
Control Rod Weight	206 lb
Maximum horizontal acceleration (SSE)	[1.2 g]
Maximum vertical acceleration (SSE)	[1.0 g]

2. DESIGNER'S ALTERNATIVES

The following alternative is available:

2.1 Perform seismic qualification of subassemblies in accordance with IEEE-323 and rely on analysis to predict response of the control assembly.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The complete neutron control assembly will be qualified for performance of its safety functions under seismic simulation. The selected approach provides a higher level of confidence than Alternative 2.1.

4. SCHEDULE REQUIREMENTS

Completion six months prior to finish of plant final design phase (3/93).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Fallback position is Alternative 2.1. However, seismic qualifications of subassemblies may increase risk of licensing delays.

Edwin C. Harvey 3/25/87
Originator Date

R F Turner 3/25/87
Department Manager Date

for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

GUIDE TUBES FLOW INDUCED VIBRATION DESIGN VERIFICATION
DDN M.10.12.09
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The effects of flow induced vibrations on the guide tubes, plenum elements and related components, due to high velocity flow must not cause damage or excessive wear that will result in those components not being able to perform their functions over their design lives.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods" Assumption 2, and F3.1.1.2.1.1.2.1.1 "Control With Movable Poisons," Assumption 5: Control assembly which passes through the upper plenum does not suffer unacceptable damage or operability problems as a result of flow induced vibrations.

1.2 Current Data Base Summary

British reactors have demonstrated that there is a potential for vibration damage of components in gas streams. Much work has been done on analytical evaluation of induced vibrations in gas streams. This general work will be used to analyze the proposed design, but the analysis should be verified by specific experimental data.

1.3 Data Needed

Data are needed on the frequency and magnitude of significant vibrations of the guide tubes, plenum elements and related components. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Core Inlet Pressure (psia)	924
Core Inlet Temperature (°F)	497
Helium Flow Past Outer Guide Tube (lb/h)	51,950
Flow Velocities (ft/s)	[75]

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Depend only on results of flow induced vibration analysis.

2.2 Small scale tests with data extrapolation to full size unit.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to test a full scale air flow mock-up of a group of guide tubes, plenum elements and related components. The test will allow interaction of the components to be evaluated. Testing will be at the upper core plenum inlet flow rates with various orientations of the components in the flow stream. Results will be extrapolated to reactor conditions as required by the modelling selected. Alternatives 2.1 and 2.2 were rejected because they cause significant increases in design uncertainty.

4. SCHEDULE REQUIREMENTS

Data from this test are required six months prior to finish of the plant final design phase (3/93).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to rely on analysis to verify the acceptability of the design. This design approach might lead to a design with excessive wear with the consequential requirement for early replacements or design retrofits.

Edwin C. Harvey 3/25/87
Originator Date

R J Turner 3/25/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/25/87

DETERMINE TYPE OF IN-CORE FLUX MONITOR UNIT
DDN M.10.12.10
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Axial neutron flux distribution data is required to monitor core performance. The purpose of this DDN is to compare various candidate sensing methods in an HTGR operating environment.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.3.2 "Monitor Neutron Flux Distribution"/Assumption 2: Testing underway between small fission chambers and gamma thermometers show that gamma thermometers are suitable for HTGR in-core flux monitoring. The ability of the gamma thermometers to measure neutrons under HTGR conditions needs to be confirmed.

1.2 Current Data Base Summary

Studies to date have indicated the most likely candidates for successful mapping of axial nuclear flux in an HTGR environment are a TOSHIBA fission chamber or a gamma thermometer.

The TOSHIBA fission chamber has been successfully tested at high temperature in the TRIGA reactor. The purpose of the TRIGA test was to prove the sensor could operate properly at low radiation levels (10^{12} n/cm^2-s) and high temperatures. (The detector shows good linearity up to about 825°C above which signal degradation occurs).

A second sensor called a gamma thermometer that measures the amount of gamma and fast neutron radiation generated in the fuel, in the form of heat, has excellent potential for success because of its simplicity. Testing to date on gamma thermometers is limited to LWR use.

1.3 Data Needed

Comparative data for the two candidate detectors is needed to verify the following:

a. Acceptable operation at HTGR reactor core/reflector conditions.

b. The operating voltage selected for the fission chamber is valid for high radiation fields.

c. Gamma heating is not a problem for the fission chamber.

d. The change in sensitivity due to burn-up of the fission chamber is not significant over the required life of the detector.

e. Exposure of the detector leadwire to high temperatures does not significantly affect operation.

The present plans are to obtain this data by installing the detectors in the Fort St. Vrain core.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

Data Parameters:

Signal versus plant power level
Signal degradation versus time (signal to noise ratio)
Effects of temperature on signal

Service Conditions:

Neutron Flux:	10^{14} n/cm^2-s
Gamma Flux:	(1.9 x 10^{11} mrem/h)
Temperature:	[1400]°F
Operating Environment:	Helium
Pressure:	800-1200 psig
Moisture:	[<10 ppm]

2. DESIGNER'S ALTERNATIVES

2.1 Select and install a detector without the benefit of verification in an HTGR environment. This would necessitate more reliance on the analytical calculations of flux distribution.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A detector evaluation program consisting of comparison of a TOSHIBA fission chamber and a gamma thermometer has been recommended because considerable risk of encountering measurement difficulties is involved in <u>not</u> verifying detector performance in an HTGR environment.

4. SCHEDULE REQUIREMENTS

The data will be needed at the beginning of the preliminary design phase (9/87) because choice of type of detection sensor impacts the reactor vessel, core, mechanisms, and handling tool design.

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to select and install a detector without the benefit of verification in an HTGR environment. This would necessitate placing more reliance on analytical calculation of the flux distribution and could necessitate a more conservative core design.

The consequences of nonexecution may result in a nonoperable system. Lack of axial power shape data could lead to core operation in a fashion detrimental to the core (excessive fuel failure, etc.).

L. M. Weaver 3/12/87
Originator Date

031387
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/25/87

VERIFY CONTROL ROD INSTRUMENTATION & CONTROL
DDN M.10.12.11
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Programmable digital controllers in conjunction with brushless DC torque motor drive amplifiers are utilized in the control rod control system. The purpose of this DDN is to verify the controller programming and ensure operational compatibility of the controlling drive amplifiers and control rod drive mechanisms.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.3 "Move Control Rods"/Assumption 3: The rod drive selection and control circuitry operation will be verified.

1.2 Current Data Base Summary

The use of programmable digital controllers and variable speed motor control for a rod control system such as that of the 4 x 350 MW(t) is a new application configuration. This application in a nuclear plant has a potential impact on plant safety if selection, direction and speed control is inadequately controlled.

1.3 Data Needed

1. The use of solid state programmable controllers in conjunction with the drive motor speed control circuitry needs verification to demonstrate:

   Drive selection capability (individual/group/automatic and manual).
   Position indication selection capability.
   Direction control selection.
   Drive speed control.
   Maximum withdrawal speed.

2. Performance characteristics (position versus time) and system response time of the control rod control system.

3. Design verification that the control rod control is compatible with control rod drive mechanisms.

Quality Assurance must be in accordance with QAL II.

1.4 <u>Data Parameters/Service Conditions</u>

<u>Operating Parameters</u>:

Drive selection capability: [Individual, Group, Sequence, Automatic Shim]
Direction selection capability: [Insert-Withdraw]
Drive speed: [2.2] in./s.
Drive stroke: [366] in.
Control response time: [TBD]

<u>Operating Environment:</u>

Reactor Building Equipment:
Temperature: [104°F]
Pressure: Atmospheric
Humidity: [90% RH]
Radiation: Nil
Other:* [None]

*Includes vibration, EMI, RFI, gas composition, etc.

2. DESIGNER'S ALTERNATIVES

2.1 Downgrade degree of automation of rod control system and use more conventional relay and selection techniques and install system without benefit of interface verification.

2.2 Install system without benefit of verification.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A programmable controller technique was selected to control the reactor control rod mechanisms because:

1. It better fits the microprocessor based distributed control scheme selected for the plant instrumentation and controls.

2. It is more reliable than conventional relay and switching mechanisms as it contains fewer moving parts.

3. It requires less physical space.

4. It requires less maintenance.

5. Significant operating flexibility is obtainable. Rods selected to be in one bank may be changed easily by reprogramming the controller for various core loadings.

A verification program has been selected to utilize the benefits of a more advanced design relative to capability for more complete automation and circuit flexibility as opposed to the use of more conventional relay/selection's techniques. The verification program also assures proper operation of the rod control circuitry prior to plant startup, thereby reducing the risk of plant startup delays.

4. SCHEDULE REQUIREMENTS

The data will be needed one year after the start of plant preliminary design (9/88).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Use programmable controllers for control rod control without design verification testing.

Without design verification the use of programmable digital controllers and variable speed control could result in unavailability of performance data when needed and impacts mechanical design of rod drive systems, plant control system, and control system design. Additional plant startup delays could be encountered should adverse equipment interface problems or performance characteristics be uncovered during plant startup.

[signature] 3/13/87
Originator — Date

[signature] 031387
Department Manager — Date

[signature] for G.R. Bramblett 3.25.87
Manager, Project Operations — Date

DATE: 2/25/87

VERIFY STARTUP NEUTRON DETECTOR & CABLING
DDN M.10.12.12
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The source level neutron monitoring requires the use of in-vessel neutron monitoring sensors (detectors). The purpose of this DDN is to confirm operability of a neutron detector and associated cabling in an HTGR environment.

1.1 Summary of/Function/Title Assumptions

F1.4.1.2.3.1.1 "Monitor Source Range Neutrons"/Assumption 4: Neutron detection can be made in-vessel in an HTGR environment for startup range power levels.

1.2 Current Data Base Summary

The current data base for low level in-vessel neutron detectors (and cabling) capable of use in HTGR startup range neutron detection dates back to prior efforts on the LHTGR. Other more current data on high temperature, high pressure neutron detectors is available from neutron detector manufacturers for LWR (primarily BWR) applications.

1.3 Data Needed

Data confirming operability, neutron sensitivity and detector/cable life at in-reactor HTGR conditions.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

Data Parameters: Neutron sensitivity versus time.
Signal to noise ratio versus time, versus temperature, and versus power.

Service Conditions:

Pressure:	[1041] psia
Temperature:	
At Detector:	[600]°F
At Cable Splice:	[500]°F
Gamma Field:	[10^8] R/h
Neutron Field:	[10-10^2] nv/cm^2-s
Atmosphere:	Reactor grade helium

2. DESIGNER'S ALTERNATIVES

2.1 Select and install neutron detectors (and cabling) without benefit of verification of operability at HTGR conditions.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A verification program is recommended to confirm operability at HTGR conditions because neutron detection for use in reactor startup is an important measurement and in-vessel neutron measurements are difficult.

4. SCHEDULE REQUIREMENTS

This data will be needed about midpoint in the preliminary design (9/88) as otherwise it can impact various interfaces with the reactor vessel/ core support, etc.

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to use in-vessel neutron detection without the benefit of verification of operability at HTGR conditions. Failure of adequate verification could cause startup delays, adversely impact plant availability (by requiring frequent replacement or longer restart times) and might require changes in installed equipment.

ABZ [signature] 3/12/87
Originator Date

[signature] 031387
Department Manager Date

[signature] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

UNIAXIAL STRENGTH DATA BASE FOR CORE SUPPORT GRAPHITE
DDN M.10.17.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite core support (CS) structure is designed to meet the stress limits of the ASME Code, Section III, Div. 2, Subsection CE. The code gives the stress limits as a percentage of the "minimum ultimate strength" which is defined in probabilistic terms (99% survivability at a confidence level of 95%).

Associated data needs: DDN M.10.17.11.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support."

Assumption 3: Grade 2020 graphite can be manufactured in the size needed for the core support structure with minimum ultimate strengths of 2400 psi in tension and 3000 psi in compression.

F3.1.1.2.1.1.2.1.1.3 "Maintain Controllable Geometry," Assumption: Grade 2020 graphite can be manufactured in the sizes needed for the core support structure with minimum ultimate strength of 2400 psi in tension and 3000 psi in compression.

1.2 Current Data Base Summary

The current reference material is 2020 graphite. Uniaxial strength (tensile, compressive, and flexural) data has been obtained in air at room temperature on axial and radial specimens from 49 standard-production billets, 10 in. in diameter and 78 in. long. A few strength measurements have been made on standard-production 2020 graphite at temperatures up to 1500°C in an inert atmosphere.

A purified grade of 2020 graphite has been investigated to improve corrosion resistance. For the purified grade 2020 graphite, uniaxial strength measurements have been made in air at ambient temperature on axial and radial specimens from two standard-production billets and one large rectangular billet of dimensions 26 in. x 26 in. x 39 in..

The current data base is judged adequate for conceptual design but needs to be increased for preliminary and final design.

1.3 Data Needed

A uniaxial strength data base sufficient to meet the ASME Code statistical requirements is needed. Quality assurance must be in accordance with the requirements for Quality Assurance Level I. The data must be valid for 2020 graphite in two billet sizes:

a. Small cylindrical billet, 7 in. in diameter and 48 in. long for core support posts.

b. Large cylindrical billet, 17 in. in diameter and 48 in. long for core support blocks.

The data base must include data on:

a. Dependence on orientation, location in billet.

b. Variation from billet to billet and lot to lot.

The full statistical data base is needed at room temperature in air. Some additional data points are needed to determine the effects of service temperatures and of the pressurized helium environment. (Note that irradiation effects are covered by DDN M.10.17.11.)

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	2400
Compressive	3000

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	700/1290

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Radiation environment

Maximum fast neutron fluence (E > 29 fJ, HTGR)
$[1 \times 10^{20}]$ n/cm

2. DESIGNER'S ALTERNATIVES

The following alternative has been considered:

2.1 Use the exisiting incomplete data base and include sufficient design margin to account for the uncertainties.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain a statistically significant uniaxial strength data base for 2020 graphite in the billet sizes used for the core support components.

Design alternative 2.1 was rejected because it would result in larger structural cross sections.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and the final data by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used resulting in larger structural sizes. This could cause additional primary coolant loop pressure drop which would reduce plant operating efficiency and increase operating costs. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of limited data.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for G.E. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

UNIAXIAL STRENGTH DATA BASE FOR PERMANENT REFLECTOR GRAPHITE
DDN M.10.17.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The permanent side reflector (PSR) is designed to meet the stress limits of the ASME Code, Section III, Div. 2, Subsection CE. The code gives the stress limits as a percentage of the "minimum ultimate strength" which is defined in probabilistic terms (99% survivability at a confidence level of 95%).

Associated data needs: DDN M.10.17.12.

1.1 Summary of Function/Title/Assumptions

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflectors."

Assumption 3: Grade 2020 graphite can be manufactured in the size needed for the permanent reflector components with minimum ultimate strengths of 1950 psi in tension and 2400 psi in compression.

1.2 Current Data Base Summary

The current reference material is 2020 graphite. Uniaxial strength (tensile, compressive, and flexural) data has been obtained in air at room temperature on axial and radial specimens from 49 standard-production billets, 10 in. in diameter and 78 in. long. A few strength measurements have been made on standard-production 2020 graphite at temperatures up to 1500°C in an inert atmosphere.

A purified grade of 2020 graphite has been investigated to improve corrosion resistance. For the purified grade 2020 graphite, uniaxial strength measurements have been made in air at ambient temperature on axial and radial specimens from two standard-production billets and one large rectangular billet of dimensions 26 in. x 26 in. x 39 in.

The current data base is judged adequate for conceptual design but needs to be increased for preliminary and final design.

1.3 Data Needed

A uniaxial strength data base sufficient to meet the ASME Code statistical requirements is needed. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

The data base must include data on:

a. Dependence on orientation and location in billet.

b. Variation from billet to billet and lot to lot.

The full statistical data base is needed at room temperature only. Some additional data points are needed to determine the effects of the service temperatures. (Note that irradiation effects are covered by DDN M.10.17.12.)

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	1950
Compressive	2400

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	900/1650

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Radiation environment

Maximum fast neutron fluence (E > 29 fJ, HTGR) $[2 \times 10^{20}]$ n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternative has been considered:

2.1 Use the existing incomplete data base and include sufficient design margin to account for the uncertainties.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain a statistically significant uniaxial strength data base for 2020 graphite in the billet size needed for the permanent side reflector blocks. This would allow the components to be designed with confidence for the requirements of the ASME code.

Design alternative 2.1 was rejected because it could result in an uneconomical design.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and the final data by [10/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used which could result in an uneconomical design. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of limited data.

KEV

W. Gorholt 3/16/87
Originator Date

R.J. Turner 3/16/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

MULTIAXIAL STRENGTH OF GRAPHITE FOR CORE SUPPORT
DDN M.10.17.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The conceptual design of the core support structure has been done on the basis of the maximum stress failure theory which is a simplified approximation whose uncertainty needs to be quantified. If the uncertainty is large, a more accurate theory will then be developed.

Associated data needs: DDN M.10.17.01.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support." Assumption 2. The maximum stress failure theory is a reasonable approximation for 2020 graphite under a multiaxial state of stress.

1.2 Current Data Base Summary

Exploratory biaxial stress tests were performed in 1980 on core support graphite. The tests yielded a limited number of biaxial stress data points. These are not sufficient to quantify the error in the maximum stress failure theory.

1.3 Data Needed

Data are needed to determine the reduction in the uniaxial strength of core support graphite due to multiaxial stress conditions. The data are needed for bi- and triaxial tension and tension/compression combinations. The data base must be adequate to show with [95]% confidence that the mean value of the uniaxial strength is not reduced by more than:

[15]% in a biaxial stress field
[20]% in a triaxial stress field

The above statistical data base is needed only for unirradiated graphite at room temperature in air. An additional small number of data points are needed on the effects of the service conditions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

	Tensile	Compressive
a. Specified minimum ultimate strength, psi		
Small cylindrical billet 7 in. diameter x 48 in. long for core support posts	2400	3000
Large cylindrical billet 17 in. diameter x 48 in. long for core support blocks	2400	3000
b. Maximum point stress, psi		
In core support post	800	1000
In core support block	800	1000
c. Service temperature range, °C/°F		
Minimum	120/248	
Maximum	700/1290	
d. Operating environment		
Primary coolant	Helium	
Pressure range	1 to 63 atmos	
e. Radiation environment		
Maximum fast fluence (E > 29 fJ, HTGR)	[1 x 20^{20}] n/cm^2	

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Continue to use the maximum stress failure theory without further validation.

2.2 Use the maximum stress failure theory, estimate the error on the basis of the existing data and include sufficient design margin to account for the uncertainties.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain a multiaxial strength data base sufficient to bound the error in the simple maximum stress theory. If the error is unacceptably large, a more accurate failure theory will then be developed.

Alternative 2.1 was rejected due to the risk of not being acceptable for licensing.

Alternative 2.2 would result in unnecessarily large structural cross sections.

4. SCHEDULE REQUIREMENTS

Preliminary data are needed by [3/89], six months prior to PSAR submittal and the final data [10/91] two years after the start of final design.

5. PRIORITY

Urgency: 3
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Designer's alternative 2.2 would be used. The consequences would be increased structural sizes. This could cause additional primary coolant loop pressure drop which would reduce plant operating efficiency and increase operating costs. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of limited data.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

MULTIAXIAL STRENGTH OF GRAPHITE FOR PERMANENT REFLECTOR
DDN M.10.17.04
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The conceptual design of the permanent side reflector has been done on the basis of the maximum stress failure theory which is a simplified approximation whose uncertainty needs to be quantified. If the uncertainty is large, a more accurate theory will then be developed.

Associated data needs: DDN M.10.17.02.

1.1 Summary of Function/Title/Assumptions

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflector." Assumption 2. The maximum stress failure theory is a reasonable approximation for 2020 graphite under a multiaxial state of stress.

1.2 Current Data Base Summary

Exploratory biaxial stress tests were performed in 1980 on permanent reflector graphite. The tests yielded a limited number of biaxial stress data points. These are not sufficient to quantify the error in the maximum stress failure theory.

1.3 Data Needed

Data are needed to determine the reduction in the uniaxial strength of permanent reflector graphite due to multiaxial stress conditions. The data are needed for bi- and triaxial tension and tension/compression combinations. The data base must be adequate to show with [95]% confidence that the mean value of the uniaxial strength is not reduced by more than:

[15]% in a biaxial stress field
[20]% in a triaxial stress field

The data base must be valid for 2020 graphite in a large rectangular billet of dimensions 20.5 in. x 20.5 in. x 39 in.. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

The above statistical data base is needed only for unirradiated graphite at room temperature in air. An additional small number of data points are needed on the effects of the service conditions.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	1950
Compressive	2400

b. Maximum point stress in PSR, psi

Tensile	650
Compressive	800

c. Service temperature range, °C/°F

Minimum	120/248
Maximum	900/1650

d. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

e. Radiation environment

Fast neutron fluence (E > 29 fJ, HTGR)	[2 x 10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Continue to use the maximum stress failure theory without further validation.

2.2 Use the maximum stress failure theory, estimate the error on the basis of the existing data and include sufficient design margin to account for the uncertainties.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain a multiaxial strength data base sufficient to bound the error in the simple maximum stress theory. If the error is unacceptably large, a more accurate failure theory will then be developed.

Alternative 2.1 was rejected due to the risk of not being acceptable for licensing.

Alternative 2.2 would in an uneconomical design.

4. SCHEDULE REQUIREMENTS

Preliminary data are needed by [3/89], six months prior to PSAR submittal and the final data by [10/91] two years after the start of final design.

5. PRIORITY

Urgency: 3
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Designer's alternative 2.2 would be used. The consequences would be an uneconomical design. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of limited data.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G.R. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

FATIGUE STRENGTH OF GRAPHITE FOR CORE SUPPORT COMPONENTS
DDN M.10.17.05
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To establish structural integrity of the core supports under cyclic loadings (e.g., plant transients, flow-induced vibration, and seismic vibration), fatigue analysis is required by the ASME Code. For this analysis, the fatigue strengths of core support component graphites must be determined.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2 "Maintain Integrity of Graphite Core Support."
Assumption 12: The cycle fatigue endurance limits for 2020 graphite specified in the Graphite Design Data Manual are valid.

1.2 Current Data Base Summary

Some uniaxial push-pull fatigue preliminary tests in air at ambient temperature have been made on axial and radial specimens of standard production grade 2020 graphite. In each case the tests were made on specimens from a single billet. The stress ratio, R (ratio between the minimum stress and the maximum stress during a cycle), varied between -2 and 0. Forty or fifty specimens were tested for each orientation and stress ratio, to a maximum of 10^5 cycles.

1.3 Data Needed

A fatigue strength data base sufficient to construct a Design Fatigue Diagram* is needed. The data base must be sufficient to establish a [95]% confidence that the mean values of the data base do not differ from the mean values of the population by more than [10]%. The data base must include:

a) Up to 10^5 cycles.

b) Stress ratio, R (ratio between the minimum and maximum stress during a cycle) ranging from [-1] to [+1].

c) Dependence on orientation and location in billet and on variation from billet to billet.

*As defined in the draft of Subsection CE of the ASME Code.

In addition, a small number of data points are needed to determine the effects of the operating environment. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, tensile/compressive, psi

Small cylindrical billet for core support posts:	2400/3000
Large cylindrical billet for core support blocks:	2400/3000

b. Data range from 1 cycle to 10^5 cycles

c. Operating Environment

Primary coolant	Helium
Pressure	1 to 63 atms

d. Service Temperature Range, °C/°F

Minimum	120/248
Maximum	700/1290

e. Irradiation

Maximum fast fluence (E > 29fJ, HTGR)
[1×10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

2.1. Use currently available fatigue data of standard production 2020 graphite and include additional design margin to cover uncertainties.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to complete the fatigue strength data base for grade 2020 graphite billets including data on the effects of the operating environment. This will allow the components to be sized with confidence to the requirements of the ASME Code.

4. SCHEDULE REQUIREMENTS

Preliminary data are needed by [10/89], at the start of final design and final data by [10/91] two years after start of final design.

5. PRIORITY

Urgency: 3
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design Alternative 2.1 would be used. The resulting increased structural cross section sizes could cause additional primary coolant loop pressure drop reducing plant operating efficiency and increasing operating costs. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of limited data.

KEU

W. Gorholt 3/10/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

FATIGUE STRENGTH OF GRAPHITE FOR PERMANENT REFLECTORS
DDN M.10.17.06
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To establish structural integrity of the permanent side reflectors under cyclic loadings (e.g., plant transients, flow-induced vibration, and seismic vibration), fatigue analysis is required by the ASME Code. For this analysis, the fatigue strengths of the permanent reflector component graphite must be determined.

1.1 Summary of Function/Title/Assumptions

Function F.1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflector," Assumption 12: The cycle fatigue endurance limits for 2020 graphite specified in the Graphite Design Data Manual are valid.

1.2 Current Data Base Summary

Some uniaxial push-pull preliminary fatigue tests in air at ambient temperature have been made on axial and radial specimens of standard production grade 2020 graphite. In each case the tests were made on specimens from a single billet. The stress ratio, R (ratio between the minimum stress and the maximum stress during a cycle), varied between -2 and 0. Forty or fifty specimens were tested for each orientation and stress ratio to a maximum of 10^5 cycles.

1.3 Data Needed

A fatigue strength data base sufficient to construct a Design Fatigue Diagram* is needed. The data base must be sufficient to establish a [95]% confidence that the mean values of the data base do not differ from the mean values of the population by more than [10]%. The data base must include:

a) Up to 10^5 cycles.

b) Stress ratio, R (ratio between the minimum and maximum stress during a cycle) ranging from [-1] to [+1].

*As defined in the draft of Subsection CE of the ASME Code.

c) Dependence on orientation and location of billet and on variation from billet to billet.

In addition, a small number of data points are needed to determine the effects of the operating environment. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	1950
Compressive	2400

b. Data range from 1 cycle to 10^5 cycles.

c. Operating environment

Primary coolant	Helium
Pressure	1 to 63 atmos

d. Service temperature range, °C/°F

Minimum	120/248
Maximum	900/1650

f. Irradiation

Maximum fast fluence (E > 29 fJ, HTGR) [2×10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternative is available:

2.1 Use currently available fatigue data of standard production 2020 graphite and include additional design margin to cover uncertainties.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to complete the fatigue strength data base for grade 2020 graphite billets including data on the effects of the operating environment. This will allow the components to be designed with confidence to the requirements of the ASME Code.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/89], at the start of final design, and final data by [10/91] two years after start of final design.

5. PRIORITY

Urgency: 3
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design Alternative 2.1 would be used, resulting in an uneconomical design. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of limited data.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for G. E. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

MINER'S LAW FOR GRAPHITE FOR CORE SUPPORT COMPONENTS
DDN M.10.17.07
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Miner's rule is used in the graphite-component fatigue analysis for combining the fatigue damage from different stress amplitudes. Miner's rule is used in metallic structures but has not been validated for graphite.

Associated data needs: DDN M.10.17.05.

1.1 Summary of Function/Title/Assumptions

Function F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support," Assumption 8: Miner's rule for estimating the cumulative fatigue is applicable to 2020 graphite.

1.2 Current Data Base Summary

No data is available on the applicability of Miner's rule to graphite.

1.3 Data Needed

Data are needed on the cumulative fatigue strength of core support graphite subjected to sequential series of stress cycles with different amplitude. The number of cases (i.e., combinations of stress amplitudes) shall be selected such that a valid comparison can be made between the measured cumulative fatigue life and the cumulative fatigue life predicted by applying Miner's rule to the constant amplitude fatigue data established in DDN M.10.17.05.

A sufficient data base is needed to determine the difference between constant amplitude fatigue life and varying amplitude fatigue life with [95]% confidence. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, tensile/compressive, psi

Small cylindrical billet for core support posts	[2400]/[3000]
Large cylindrical billet for core support blocks	[2400]/[3000]

b. Data range from 1 cycle to 10^5 cycles with stress amplitudes of [Later].

c. Operating environment

Primary coolant	Helium
Pressure	1 to 63 atmos

d. Service temperature range, °C/°F

Minimum	120/248
Maximum	700/1290

e. Irradiation

Maximum fast fluence (E > 29 fJ, HTGR) $[1 \times 10^{20}]$ n/cm^2

2. DESIGNER'S ALTERNATIVES

2.1 Assume Miner's rule to be applicable without validation and design the core support components on this basis.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain data on cumulative fatigue damage and verify Miner's rule for grade 2020 graphite at room temperatures. The selected approach would allow the structural elements to be sized with confidence to the requirements of the ASME Code reducing the risk of licensing difficulties.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [3/89], six months prior to PSAR submittal and final data by [10/91] two years after start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: H
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative 2.1 would be used. The consequences would be licensing difficulties in trying to convince the NRC that Miner's rule can safely be used for graphite in spite of the lack of validation.

If the data should show Miner's rule is not suitable for graphite, another design rule must be found. In that event additional testing and theoretical studies may be necessary.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

MINER'S LAW FOR GRAPHITE FOR PERMANENT SIDE REFLECTORS
DDN M.10.17.08
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Miner's rule is used in the graphite-component fatigue analysis for combining the fatigue damage from different stress amplitudes. Miner's rule is used in metallic structures but has not been validated for graphite.

Associated data needs: DDN M.10.17.06.

1.1 Summary of Function/Title/Assumptions

Function F.1.1.1.1.2.2.1.3.2 "Maintain Integrity of Side Reflector."
Assumption 8: Miner's rule for estimating the cumulative fatigue is applicable to 2020 graphite.

1.2 Current Data Base Summary

No data is available on the applicability of Miner's rule to graphite.

1.3 Data Needed

Data are needed on the cumulative fatigue strength of permanent reflector graphite subjected to sequential series of stress cycles with different amplitude. The number of cases (i.e., combinations of stress amplitudes) shall be selected such that a valid comparison can be made between the measured cumulative fatigue life and the cumulative fatigue life predicted by applying Miner's rule to the constant amplitude fatigue data established in DDN M.10.10.06.

A sufficient data base is needed to determine the difference between constant amplitude fatigue life and varying amplitude fatigue life with [95]% confidence. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	1950
Compressive	2400

b. Data range from 1 cycle to 10^5 cycles with stress amplitudes of [Later]

c. Operating Environment

Primary coolant	Helium
Pressure	1 to 63 atms

d. Service Temperature Range, °C/°F

Minimum	120/218
Maximum	900/1650

e. Irradiation

Maximum fast fluence (E > 29 fJ, HTGR) [2×10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

2.1 Assume Miner's rule to be applicable without validation and design the permanent reflector components on this basis.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain data on cumulative fatigue damage and verify Miner's rule for grade 2020 graphite at room temperature. The selected approach would allow the structural elements to be sized with confidence to the requirements of the ASME Code reducing the risk of licensing difficulties.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [3/89], six months prior to PSAR submittal and final data by [10/91] two years after start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: H
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 2.1 would be used. The consequences would be licensing difficulties in trying to convince the NRC that Miner's rule can safely be used for graphite in spite of the lack of validation.

If the data should show that Miner's rule is not suitable for graphite, another design rule must be found. In that event additional testing and theoretical studies may be necessary.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

ELASTIC PROPERTIES DATA BASE FOR CORE SUPPORT GRAPHITE
DDN M.10.17.09
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite core support (CS) structure is designed to meet the allowable stress limits specified by the ASME Code Section III, Div. 2, Subsection CE. The code requires an adequate data base for the elastic properties.

Associated data needs: DDN M.10.17.11.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support."

Assumption 3: The mean values for the elastic properties of Young's modulus and Poisson's ratio given in the Graphite Design Data Manual for 2020 graphite are valid.

1.2 Current Data Base Summary

The current reference material is 2020 graphite. Uniaxial tensile and compressive Young's moduli data has been obtained in air at room temperature on axial and radial specimens from 49 standard-production billets, 10 in. in diameter and 78 in. long. A few measurements have been made of Young's modulus at elevated temperatures.

A purified grade of 2020 graphite has been investigated to improve corrosion resistance. For the purified grade 2020 graphite, Young's modulus measurements have been made in air at ambient temperature on axial and radial specimens from two standard-production billets and one large rectangular billet of dimensions 26 in. x 26 in. x 39 in.

The current data base is judged adequate for conceptual design but needs to be increased for preliminary and final design.

1.3 Data Needed

The required data must be valid for 2020 graphite in two different sizes of billets:

a. Small cylindrical billet, 7 in. in diameter and 48 in. long for core support posts.

b. Large cylindrical billet, 17 in. in diameter and 48 in. long for core support blocks.

The data base must be sufficient to establish the mean values of Young's Modulus and Poisson's Ratio within ± [10]% and ± [25]%, respectively, at [95]% confidence. Some additional data points are needed to determine the effects of the operating environment (Note that irradiation effects are covered under DDN M.10.17.11). Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	2400
Compressive	3000

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	700/1290

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Radiation environment

Fast neutron fluence (E > 29 fJ, HTGR) [1×10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternative has been considered:

2.1 Complete the design on the basis of the currently available data.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to complete the Young's modulus and Poissons's ratio data base on grade 2020 graphite including data to reflect the conditions expected in a modular HTGR to reduce uncertainties in the predicted structural response of the core support structure.

The use of design alternative 2.1 would result in a weakened licensing position.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and the final data by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used. The consequences would be a risk of rejection during licensing which could result in a crash technology program and possibly schedule delays.

KEV

W. Gorholt 3/16/87
Originator Date

R. Turner 3/16/87
Department Manager Date

for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

ELASTIC PROPERTIES DATA BASE FOR PERMANENT REFLECTOR GRAPHITE
DDN M.10.17.10
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite permanent side reflectors (PSR) are designed to meet the allowable stress limits specified by the ASME Code Section III, Div. 2, Subsection CE. The code requires an adequate data base for the elastic properties.

Associated data needs: DDN M.10.17.12.

1.1 Summary of Function/Title/Assumptions

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflector."

Assumption 3: The mean values for the elastic properties of Young's modulus and Poisson's ratio given in the Graphite Design Data Manual for 2020 graphite are valid.

1.2 Current Data Base Summary

The current reference material is 2020 graphite. Uniaxial tensile and compressive Young's modulus data has been obtained in air at room temperature on axial and radial specimens from 49 standard-production billets, 10 in. in diameter and 78 in. long. A few Young's modulus measurements have been made at temperatures up to 1500°C. No measurements have been made from large billets.

A purified grade of 2020 graphite has been investigated to improve corrosion resistance. For the purified grade 2020 graphite, Young's modulus measurements have been made in air at ambient temperature on axial and radial specimens from two standard-production billets and one large rectangular billet of dimensions 26 in. x 26 in. x 39 in.

Current data base is judged adequate for conceptual design but needs to be increased for preliminary and final design, especially in the large sizes appropriate to the permanent reflectors.

1.3 Data Needed

A Young's modulus and Poisson's ratio data base is required. The data base must be valid for 2020 graphite in the 20.5 in. x 20.5 in. x 39 in. billet selected for the permanent side reflector blocks. The data base must be sufficient to establish the mean values of Young's Modulus and Poisson's Ratio within ± [10]% and ± [25]%, respectively, at [95]% confidence. Some additional data points are needed to determine the effects of the operating environment. (Note that irradiation effects are covered under DDN M.10.17.12.) Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

Tensile	[1950]
Compressive	[2400]

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	900/1650

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Radiation environment

Fast neutron fluence (E > 29 fJ, HTGR) [2 x 10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternative has been considered:

2.1 Complete the design on the basis of the currently available data.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to complete the Young's modulus and Poisson's ratio data base on grade 2020 graphite including data to reflect the conditions expected in a modular HTGR to reduce uncertainties in the predicted structural response of the permanent side reflector.

The use of design alternate 2.1 would result in a weakened licensing position.

4. SCHEDULE REQUIRMENTS

Preliminary date are needed by [10/88], one year after the start of preliminary design and the final data by [10/91], two years after the start of final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Design alternative 2.1 would be used. The consequences would be a risk of rejection during licensing which could result in a crash technology program and possibly schedule delays.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

IRRADIATION EFFECTS ON MECHANICAL PROPERTIES OF CORE SUPPORT GRAPHITE
DDN M.10.17.11
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite core support components are required to satisfy the stress limits specified by the ASME Code Section III, Div. 2, Subsection CE. In showing compliance with the stress limits, the effects of the low level irradiation must be included.

Associated data needs are DDNs: M.10.17.01, M.10.17.09.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support."

Assumption 5: The effects of radiation on 2020 graphite mechanical properties are negligible.

1.2 Current Data Base Summary

A limited amount of irradiation test data are available on grade 2020 graphite but only at a fluence greater than 1.3×10^{21} neutrons/cm^2.

1.3 Data Needed

Data are needed to define the effect of low levels of fast fluence on the Young's modulus, Poisson's ratio, and strength of 2020 graphite. The data base must be sufficient to establish with [95]% confidence that the effects of irradiation are to 1) increase the minimum ultimate strength (see DDN M.10.17.01) and 2) change the mean values of Young's modulus and Poisson's ratio by less than [10]% (see DDN M.10.17.9). If the changes to Young's Modulus and Poisson's ratio exceed 10% the changes must be determined within ±[5]% for Young's Modulus and ±[20]% for Poisson's ratio. The required data must be valid for 2020 graphite in two sizes:

a. Small cylindrical billet, 7 in. in diameter and 48 in. long for core support posts.

b. Large cylindrical billet, 17 in. in diameter and 48 in. long for core support blocks.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

	Tensile	Compressive
Small cylindrical billet for core support posts	[2400]	[3000]
Large cylindrical billet for core support blocks	[2400]	[3000]

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	700/1290

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Fast fluence range, neutrons/cm^2 (E > 29 fJ, HTGR)

Minimum	0
Maximum	[1 x 10^{20}]

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Complete the design on the assumption that the irradiation effects are negligible without further validation.

2.2 Increase the depth of the replaceable reflector so as to reduce the exposure of the core support structure to negligible levels of fast fluence.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to establish a data base for quantifying the effects of low level irradiation.

Design Alternative 2.1 would weaken the licensing position. An increased depth of the bottom reflector (Alternative 2.2) would lead to reduced plant efficiency through increased core pressure drop and increased capital costs through the consequential increase in the size of the core cavity.

4. SCHEDULE REQUIREMENTS

Preliminary data are needed by [3/89], six months prior to PSAR submittal and final data by [10/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used. The consequences would be a risk of rejection during licensing, which could result in a crash technology program and possible schedule delays.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

Dulisson for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

IRRADIATION EFFECTS ON MECHANICAL PROPERTIES OF PERMANENT REFLECTOR GRAPHITE
DDN M.10.17.12
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Permanent side reflectors (PSR) are required to satisfy the stress limits specified by the ASME Code Section III, Div. 2, Subsection CE. In showing compliance with the stress limits, the effects of the low level irradiation must be included.

Associated data needs: DDN's M.10.17.02 and 10.17.10.

1.1 Summary of Function/Title/Assumptions

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflector."

Assumption 5: The effects of radiation on 2020 graphite mechanical properties are negligible.

1.2 Current Data Base Summary

A limited amount of irradiation test data are available on grade 2020 graphite but only at a fluence greater than 1.3×10^{21} neutrons/cm^2.

1.3 Data Needed

Data are needed to define the effect of low levels of fast fluence on the Young's modulus, Poisson's ratio, and strength of 2020 graphite. The data base must be sufficient to establish with [95]% confidence that the effects of irradiation are to 1) increase the minimum ultimate strength (see DDN M.10.17.01) and 2) change the mean values of Young's modulus and Poisson's ratio by less than [10]% (see DDN M.10.17.10). The required data must be valid for 2020 graphite in the billet size selected for the permanent reflector blocks 20.5 in. x 20.5 in. x 39 in. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Specified minimum ultimate strength, psi

	Tensile	Compressive
	[1950]	[2400]

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	900/1650

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 65 atmos

d. Fast fluence range, neutrons/cm^2 (E > 29 fJ, HTGR)

Minimum	0
Maximum	[2 x 10^{20}]

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Complete the design on the assumption that the irradiation effects are negligible without further validation.

2.2 Increase the number of replaceable side reflectors so as to reduce the exposure of the permanent side reflectors to negligible levels of fast fluence.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to establish a data base for quantifying the effects of low level irradiation.

Design Alternative 2.1 would weaken the licensing position. Additional replaceable reflectors (Alternative 2.2) would lead to a larger core diameter and consequent increases in the diameters of the core barrel and of the reactor vessel. The consequence would be an increase in the capital cost.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [3/89], six months prior to PSAR submittal and final data are needed by [10/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used. The consequences would be a risk of rejection during licensing, which could result in a crash technology program and possible schedule delays.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

THERMAL PROPERTIES OF CORE SUPPORT GRAPHITE
DDN M.10.17.13
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To calculate the coolant and graphite temperatures and the associated size changes and thermal stresses in the core support components, a thermal properties data base is needed.

Associated data needs: DDN 10.17.15.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support."

Assumption 3: The mean values of thermal expansivity, thermal conductivity, and specific heat given in the Graphite Design Data Manual for 2020 graphite are valid.

Assumption 4: The lower bound for the emissivity of 2020 graphite is 0.8.

1.2 Current Data Base Summary

Some thermal property data (e.g. specific heat, thermal expansivity, and thermal conductivity) of Grade 2020 graphite have been obtained from a very limited number of logs (or billets). The existing data base includes the mean value of the tested population, the within-log standard deviation (for a particular orientation and location) and the log-to-log standard deviation. The uncertainty in the estimates of the population mean values and standard deviation is felt to be large because of the limited size of the data base. The current data base for grade 2020 graphite is limited to specimens from ten 7-in. diameter logs and one billet 24 in. x 24 in. x 39 in.

1.3 Data Needed

Data are needed for thermal expansivity, thermal conductivity, emissivity, and specific heat. The data base must be sufficient to establish with [95]% confidence that the mean values of the data base do not differ from the mean values of the population by more than [10]%.

The data base must be valid for two different billet sizes:

a. Small cylindrical billet, 7 in. in diameter and 48 in. long for core support posts.

b. Large cylindrical billet, 17 in. in diameter and 48 in. long for core support blocks.

The data base must include data on:

a. Dependence on orientation, location in billet.

b. Variation from lot to lot, billet to billet.

c. Effects of the reactor operating conditions (Note that irradiation effects are covered under DDN M.10.17.15.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Temperature range, °C/°F

Minimum service	120/248
Maximum service	700/1290

b. Operating environment

Primary coolant	Helium
Fast neutron fluence (E > 29 fJ, HTGR)	[1×10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use currently available data and add design margin to account for the uncertainties.

2.2 Eliminate the need for additional thermal properties by reducing the thermal stresses such that the existing data base could be supplemented with conservative assumptions without needing additional design margin.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to complete the thermal properties data base for use in the design.

Alternative 2.1 would require components of larger cross sections and could result in higher core pressure drop and increased capital and operating costs.

Alternative 2.2 would require slower shutdown transients which could only be achieved through a complete redesign of the plant control system and was thus rejected as being unrealistic.

4. SCHEDULE REQUIREMENTS

Preliminary data are needed by [10/88], one year after the start of preliminary design and final data by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used resulting in larger structural sizes. This could cause additional primary coolant loop pressure drop which would reduce plant operating efficiency and increase operating costs. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of the limited data.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for G.R. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

THERMAL PROPERTIES OF PERMANENT REFLECTOR GRAPHITE
DDN M.10.17.14
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To calculate the coolant and graphite temperatures and the associated size changes and thermal stresses in the permanent reflector components, a thermal properties data base is needed.

Associated data needs are: DDN M.10.17.16.

1.1 Summary of Function/Title/Assumptions

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflectors."

Assumption 3: The mean value of thermal expansivity, thermal conductivity, and specific heat given in the Graphite Design Data Manual for 2020 graphite are valid.

Assumption 4: The lower bound for the emissivity of 2020 graphite is 0.8.

1.2 Current Data Base Summary

Some thermal property data (e.g., specific heat, thermal expansivity, and thermal conductivity) of Grade 2020 graphite were obtained from a very limited number of logs (or billets). The existing data base includes the mean value of the tested population, the within-log standard deviation (for a particular orientation and location) and the log-to-log standard deviation. The uncertainty in the estimates of the population mean values and standard deviation is felt to be large because of the limited size of the data base. The current data base for grade 2020 graphite is limited to specimens from ten 7-in. diameter logs and one billet 24 in. x 24 in. x 39 in.

1.3 Data Needed

Data are needed for thermal expansivity, thermal conductivity, emissivity, and specific heat. The data base must be sufficient to establish with [95]% confidence that the mean values of the data base do not differ from the mean values of the population by more than [10]%.

The data base must be valid for a billet size of 20.5 in. x 20.5 in. x 39 in. and must include data on:

a. Dependence on orientation, location in billet.

b. Variation from lot to lot, billet to billet.

c. Effects of the reactor operating condition (Note that irradiation effects are covered under DDN M.10.17.16).

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Temperature range, °C/°F

Minimum service	120/248
Maximum service	900/1650

b. Operating environment

Primary coolant	Helium
Fast neutron fluence (E > 29 fJ, HTGR)	$[2 \times 10^{20}]$ n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use currently available data and add design margin to account for the uncertainties.

2.2 Eliminate the need for additional thermal properties by reducing the thermal stresses such that the existing data base could be supplemented with conservative assumptions without needing additional design margin.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to complete the thermal properties data base for use in the design.

Alternative 2.1 may require components of larger cross sections which may result in increased capital cost.

Alternative 2.2 would require the slower shutdown transients which could only be achieved through a complete redesign of the plant control system and was thus rejected as being unrealistic.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88], one year after the start of preliminary design and final data by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used. This may require larger structural sizes and could result in an increase in the diameters of the core barrel and of the reactor vessel and, consequently, higher capital cost. In addition, licensing difficulties may be encountered when trying to convince the NRC that the approach has adequate safety margin in light of the limited data.

REV

W. Gotholt 3/16/87

Originator Date

R F Turner 3/16/87

Department Manager Date

[illegible] for

G. R. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

IRRADIATION EFFECTS ON THERMAL PROPERTIES OF CORE SUPPORT GRAPHITE
DDN M.10.17.15
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To calculate the temperature gradients and the associated thermal stresses in the core support components, the effects of low level irradiation on the thermal properties must be established.

Associated data needs are DDNs: M.10.17.13.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support."

Assumption 5: The effects of radiation on 2020 graphite thermal properties are negligible.

1.2 Current Data Base Summary

There is no existing data on the effects of low levels of neutron irradiation on the thermal properties of graphite grade 2020.

1.3 Data Needed

Data are needed to define the effects of low levels of fast fluence on the thermal expansivity, thermal conductivity, and specific heat of graphite grade 2020. The data base must be sufficient to establish with [95]% confidence that the irradiation effects do not change the mean value of the above properties by more than [10]%. If the change is greater than [10]%, the change must be determined within ±[5]%. The required data base must be valid for graphite grade 2020 in two sizes:

a. Small cylindrical billets 7 in. diameter x 48 in. long for core support posts.

b. Large cylindrical billets 17 in. diameter x 48 in. long for core support blocks.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Graphite temperature range, °C/°F

Minimum service	120/248
Maximum service	700/1290

b. Operating environment

Primary coolant	Helium
Fast neutron fluence (E > 29 fJ, HTGR)	[1 x 10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

2.1 Complete the design on the basis that the irradiation effects are negligible without further validation.

2.2 Reduce the exposure of the core support structure to negligible levels of fast fluence by increasing the depth of the bottom reflector.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to establish a data base for quantifying the effects of low level irradiation.

Alternative 2.1 would weaken the licensing position.

Alternative 2.2 would increase the core cavity and also the core pressure drop and thus would result in larger capital and operating costs.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88], one year after the start of preliminary design and final data are required by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 1
Cost benefit: L
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 2.1 would be used. The consequences would be a risk of rejection during licensing, which could result in a crash technology program and possible schedule delays.

REV

W. Gorholt 3/16/87

Originator Date

R F Turner 3/16/87

Department Manager Date

for G.C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

IRRADIATION EFFECTS ON THERMAL PROPERTIES OF PERMANENT SIDE REFLECTOR GRAPHITE
DDN M.10.17.16
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To calculate the temperature gradients and the associated thermal stresses in the permanent side reflector components, the effects of low level irradiation on the thermal properties must be established.

Associated data needs are DDNs: M.10.17.14.

1.1 Summary of Function/Title/Assumptions

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflectors."

Assumption 5: The effects of radiation on 2020 graphite thermal properties are negligible.

1.2 Current Data Base Summary

There is no existing data on the effects of low levels of neutron irradiation on the thermal properties of graphite grade 2020.

1.3 Data Needed

Data are needed to define the effects of low levels of fast fluence on the thermal expansivity, thermal conductivity, and specific heat of graphite grade 2020. The data base must be sufficient to establish a [95]% confidence that the irradiation effects do not change the mean value of the above properties by more than [10]%. If the change is greater than [10]%, the change must be determined within $\pm$[5]%. The required data base must be valid for graphite grade 2020 in a billet size of dimensions 20.5 in. x 20.5 in. x 39 in. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Temperature range, °C/°F

Minimum service	[120/248]
Maximum service	[900/1650]

b. Operating environment

Primary coolant	Helium
Fast neutron flux (E > 29 fJ, HTGR)	[2 x 10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Complete the design on the basis that the irradiation effects are negligible without further validation.

2.2 Reduce the exposure of permanent side reflectors to negligible levels of fast fluence by adding replaceable reflector elements.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to establish a data base for quantifying the effects of low level irradiation.

Alternative 2.1 would weaken the licensing position.

Alternative 2.2 could increase the core cavity resulting in a larger reactor vessel and thus higher capital cost.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88], one year after the start of preliminary design and final data by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 1
Cost benefit: L
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 2.1 would be used. The consequences would be a risk of rejection during licensing, which could result in a crash technology program and possible schedule delays.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CORROSION CHARACTERISTICS OF CORE SUPPORT GRAPHITE
DDN M.10.17.17
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The core support components are designed with a corrosion allowance on the basis that the graphite corrosion due to impurities in the helium is limited to a skin effect. The amount of corrosion needs to be determined so that the adequacy of the corrosion allowance can be confirmed.

1.1 Summary of Function Number/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2 "Maintain Integrity of Graphite Core Support." Assumption 9: Graphite corrosion is limited to a maximum depth of 2 mm.

F3.1.1.2.1.1.2.1.1.3 "Maintain Controllable Geometry," Assumption: Corrosion of 2020 graphite is limited to a maximum depth of 2 mm.

1.2 Current Data Base Summary

The corrosion of core support graphite (Stackpole 2020) by coolant impurities (H_2O and O_2) may be mass-transfer limited, chemical-reaction limited, or a combination of both; consequently, both processes must be characterized. The transport of coolant impurities in core support graphite is by pore diffusion; the transport rates increase with increasing graphite burnoff. The process is characterized by an effective diffusion coefficient; the reference correlation was obtained for H_2O transport in H451 graphite with 1% burnoff.

The reference correlations for the kinetics of 2020 corrosion by coolant impurities are based upon laboratory measurements on other nuclear graphites. The reaction rate of H_2O with 2020 is taken to be three times higher than that derived for H451 fuel element graphite (see DDN M.10.18.08, Section 1.2). For oxidation of core support graphite by air, the rate expression derived from lab measurements on H327 graphite is used.

1.3 Data Needed

Data are needed to describe the corrosion of 2020 graphite due to coolant impurities. The data must be sufficient to predict the burnoff within a factor of 2 with 95% confidence, both for normal operation and for the postulated moisture and air ingress events. To the extent these are significant, the effects of low level irradiation must be included. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Maximum fast fluence (E > 29 fJ, HTGR)	[1×10^{20}] n/cm^2
Maximum gamma flux	[TBD] MeV/cm^2-s
Primary coolant temperature range	[300 - 750] °C
Graphite temperature range	[120 - 700] °C
Maximum time averaged coolant impurity levels	[2] ppm H_2O [5] ppm CO [2] ppm CO_2 [TBD] ppm O_2 [TBD] ppm H_2 Total Oxidants <[10] ppm maximum, but not to exceed [600] ppm days per year
Helium coolant pressure	1 to 63 atms

Moisture Ingress Conditions

	Maximum Concentration (ppmv)
Moisture ingress with steam generator dump failure (DBE-9)	660
Moisture ingress with moisture monitor failure (DBE-8)	18,000

	Amount of Water Leaking into Reactor Vessel
Moisture ingress without steam generator dump (SRDC-6, 7)	1820 lb

Air Ingress Condition

	Amount of Air Ingress
Depressurized conduction cooldown (SRDC-10)	21 lb-mole

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Use current data base with associated uncertainties and add margin.

2.2 Impose tighter tech specs on primary coolant oxidant levels.

2.3 Use a higher purity, more corrosion resistant graphite.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain the basic corrosion data so that calculations can be performed to confirm that the corrosion of the core support structures is limited to a skin effect. Design Alternative 2.1 is rejected because the uncertainties in the current data base would require a large corrosion allowance and might also result in licensing difficulties. Design Alternative 2.2 is rejected because imposition of tighter tech spec limits on coolant impurities is expected to adversely impact availability. Design Alternative 2.3 is rejected because development and qualification of a higher purity core support graphite would add significant development costs.

4. SCHEDULE REQUIREMENTS

Preliminary data by [3/89], six months prior to PSSAR submittal and final data by [9/92], one year prior to FSSAR submittal.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative 2.1 would be used. The acceptable thickness of the corrosion allowance is, however, limited by the dimensional requirements, and it may not be possible to add enough margin to cover all the uncertainties. The consequences are therefore a weakened licensing position in addition to cost increases resulting from larger component sizes.

REL

W. Goholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CORROSION CHARACTERISTICS OF PERMANENT REFLECTOR GRAPHITE
DDN M.10.17.18
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The permanent reflector components are designed with a corrosion allowance on the basis that the graphite corrosion due to impurities in the helium is limited to a skin effect. The amount of corrosion needs to be determined so that the adequacy of the corrosion allowance can be confirmed.

1.1 Summary of Function Number/Title/Assumptions

F1.1.1.1.2.2.1.3.2 "Maintain Integrity of Side Reflectors."

Assumption 9: Graphite corrosion is limited to a maximum depth of [2] mm.

1.2 Current Data Base Summary

The corrosion of permanent reflector graphite (Stackpole 2020) by coolant impurities (H_2O and O_2) may be mass-transfer limited, chemical-reaction limited, or a combination of both; consequently, both processes must be characterized. The transport of coolant impurities in core support graphite is by pore diffusion; the transport rates increase with increasing graphite burnoff. The process is characterized by an effective diffusion coefficient; the reference correlation was obtained for H_2O transport in H451 graphite with 1% burnoff.

The reference correlations for the kinetics of 2020 corrosion by coolant impurities are based upon laboratory measurements on other nuclear graphites. The reaction rate of H_2O with 2020 is taken to be three times higher than that derived for H451 fuel element graphite (see DDN M.10.18.08, Section 1.2). For oxidation of core support graphite by air, the rate expression derived from lab measurements on H327 graphite is used.

1.3 Data Needed

Data are needed to describe the corrosion of 2020 graphite due to coolant impurities. The data must be sufficient to predict the burnoff within a factor of 2 with 95% confidence, both for normal operation and for the postulated moisture and air ingress events. To the extent these are significant, the effects of low level irradiation must be included. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation

Environment	Helium
Maximum fast fluence (E > 29 fJ, HTGR)	[2 x 10^{20}] n/cm^2
Maximum gamma flux	[TBD] MeV/cm^2-s
Primary coolant temperature range	[300 - 750] °C
Graphite temperature range	[120 - 900] °C
Maximum time averaged coolant impurity levels	[2] ppm H_2O [5] ppm CO [2] ppm CO_2 [TBD] ppm O_2 [TBD] ppm H_2 Total Oxidants <[10] ppm maximum, but not to exceed [600] ppm days per year
Helium coolant pressure	1 to 63 atms

Moisture Ingress Conditions

	Maximum Concentration (ppmv)
Moisture ingress with steam generator dump failure (DBE-9)	660
Moisture ingress with moisture monitor failure (DBE-8)	18,000

	Amount of Water Leaking into Reactor Vessel
Moisture ingress without steam generator dump (SRDC-6, 7)	1820 lb

Air Ingress Condition

	Amount of Air Ingress
Depressurized conduction cooldown (SRDC-10)	21 lb-mole

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Use current data base with associated uncertainties and add margin.

2.2 Impose tighter tech specs on primary coolant oxidant levels.

2.3 Use a higher purity, more corrosion resistant graphite.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain the basic corrosion data so that calculations can be performed to confirm that the corrosion of 2020 graphite under normal operating and H_2O ingress conditions is limited to a skin effect. Design Alternative 2.1 is rejected because the uncertainties in the current data base would require a large corrosion allowance and might also result in licensing difficulties. Design Alternative 2.2 is rejected because imposition of tighter tech spec limits on coolant impurities is expected to adversely impact availability. Design Alternative 2.3 is rejected because development and qualification of a higher purity permanent reflector graphite would add significant development costs.

4. SCHEDULE REQUIREMENTS

Preliminary data by [3/89], six moinths prior to PSSAR submittal and final data by [9/92], one year prior to FSSAR submittal.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative 2.1 would be used. The acceptable thickness of the corrosion allowance is, however, limited by the dimensional requirements, and it may not be possible to add enough margin to cover all the uncertainties. The consequences are therefore a weakened licensing position in addition to cost increases resulting from larger component sizes.

KEV

W. Gotholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

allison for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CONFIRM LARGE SIZE GRAPHITE FOR PERMANENT REFLECTOR
DDN M.10.17.21
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The permanent side reflector (PSR) is designed to be made from large blocks of grade 2020 graphite. The ability to manufacture blocks of this size with consistent properties needs confirmation.

Associated Data Needs: DDN M.10.17.02.

1.1 Summary of Function/Title/Assumptions

F.1.1.2.1.2.1.2.2.1 "Limit Flow in Horizontal Gaps"
Assumption 2: Large grade 2020 graphite blocks are available.

1.2 Current Data Base Summary

Large billets of 2020 graphite of the size needed for the PSR blocks are not routinely produced. One large block of grade 2020 of dimensions 26 in. x 26 in. x 39 in. was produced in 1983 and was used for characterization tests at GA.

1.3 Data Needed

The process for producing graphite grade 2020 in a billet size of [0.52 m x 0.52 m x 1.00 m (20.5 in. x 20.5 in. x 39 in.)] is needed so that the ability to supply permanent side reflector blocks in sufficient quantities and with consistent properties can be ensured. The process must be such that all the billets produced have the required minimum ultimate strength (see DDN M.10.17.02). If grade 2020 in the large billet size does not have the required properties, an alternative material must be developed. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

	Tensile	Compressive
a. Specified minimum ultimate strength, psi	[1950]	[2400]

b. Service temperature range, °C/°F

Minimum	[120/248]
Maximum	[500/932]

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Fast fluence, neutrons/cm^2 (E > 29 fJ, HTGR)

Maximum	[2×10^{20}]

2. DESIGNER'S ALTERNATIVES

2.1 Decrease the size of the permanent reflector blocks to that selected for the largest core support component.

2.2 Assume that large blocks with consistent properties can be produced with an existing process.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to develop a controlled and reproducible grade 2020 graphite (or an equivalent material) in a size of 20.5 in. x 20.5 in. x 39 in. The use of smaller blocks (Alternative 2.1) would impact the efficiency of the core by increasing the bypass flow through the larger number of gaps. Capital cost would also increase because of the need to machine and install a larger number of components. Alternative 2.2 would result in a greater billet to billet strength variation than assumed in the data base and thus an unnecessarily high rejection rate.

4. SCHEDULE REQUIRMENTS

Preliminary confirmation that the large blocks can be manufactured from grade 2020 is needed at the start of preliminary design [10/87] and the final confirmation one year later [10/88].

5. PRIORITY

Urgency: 1
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design Alternative 2.1 would be used. The consequences of smaller permanent side reflector blocks would be 1) a higher capital cost due to the need to machine a larger number of blocks, and 2) a reduced plant efficiency due to more core bypass flow.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

NDE DATA FOR REACTOR INTERNALS GRAPHITE SPECIFICATIONS
DDN M.10.17.22
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Nondestructive testing techniques are required for product control during procurement of graphite for the reactor internal structures.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support." Assumption 4: The production graphite will have the same properties as the design data base.

F1.1.1.1.2.2.1.3.2, "Maintain Integrity of Side Reflectors." Assumption 4: The production graphite will have the same properties as the design data base.

1.2 Current Data Base Summary

Oak Ridge National Laboratory has developed nondestructive techniques and produced data on their accuracy and limitations using Stackpole Carbon Company's grade 2020 graphite. The data base is too small to validate these techniques so they can be used in material NDE specifications to control the product of mass produced graphite.

1.3 Data Needed

Data are needed to validate NDE techniques and write material control specifications for the procurement of graphite for reactor internal structures. The NDE techniques must be sufficiently accurate to (1) detect flaws in the largest billets used for the reactor internal structures, and (2) determine the tensile strength of smaller specimens with an error no greater than [10%]. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

NDE will be conducted at room temperature conditions.

2. DESIGNER'S ALTERNATIVES

2.1 Control the strength of the production material by destructive strength testing only. This would require destructive testing of several tensile test coupons from each billet in addition to more extensive mapping of at least one billet from each lot.

2.2 Reduce the need for product control by accepting a greater uncertainty in the strength of the production material. This would, however, result in a lower minimum ultimate strength (which is statistically defined) and hence lower allowable stresses.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to use nondestructive examination in combination with a minimum of destructive testing to ensure that the production graphite satisfies the specifications. This approach is the most cost effective since NDE is less expensive than conventional destructive testing (Alternative 2.1). Alternative 2.2 would cause structural cross sections to be larger resulting in an increased size envelope and higher primary coolant pressure drop. This would reduce plant operating efficiency and cause higher capital and operating costs.

4. SCHEDULE REQUIREMENTS

The data is needed by [9/91], before starting procurement of graphite for the permanent graphite components. This is assumed to be two years before the end of the final design phase (9/93).

5. PRIORITY

Urgency: 5
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative 2.1 would be used. The consequences will be possible higher cost of the graphite material.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for G.E. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CONFIRM STRENGTH OF GRAPHITE CORE SUPPORT
DDN M.10.17.23
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The ultimate failure load of the graphite core support structure is much higher than predicted with stress analysis. Therefore, testing is needed to determine the load capacity.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.2.2.2.2, "Maintain Integrity of Graphite Core Support." Assumption 1. Stress analysis underpredicts the load capacity.

1.2 Current Data Base Summary

Full-scale testing of FSV core support structure modules (1968) confirmed adequate safety factors against vertical loads. A series of tests on LHTGR core support posts and seats (1976 and 1977) did not correlate well with analytical predictions. Specifically, in the first tests premature failure of the seats was experienced. In the second tests (after redesign of the seats), the experimental ultimate load exceeded the analytical predictions. Subsequently, more detailed three dimensional analysis improved the correlations but resulted in a requirement to validate future designs.

1.3 Data Needed

Data are required to confirm that the ultimate load capacity of the graphite core support structure is adequate. The design to be confirmed will be that selected during conceptual and preliminary design. Data are also required on the load at which initial cracking occurs, if different from the ultimate load. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Maximum vertical design loads per core column, lb

Deadweight	[4500]
Pressure drop	[1100]
Operating basis earthquake	[4500]
Differential expansion loads	[TBD]

b. Service temperature range, °C/°F

Minimum	120/248
Maximum	700/1290

c. Operating environment

Primary coolant	Helium
Pressure range	1 to 63 atmos

d. Radiation environment

Maximum fast fluence (E > 29 fJ, HTGR)
[1×10^{20}] n/cm^2

2. DESIGNER'S ALTERNATIVES

2.1 Qualify the design on the basis of stress analysis alone.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to design the core support structure on the basis of stress analysis in combination with earlier test results and to confirm the design by validating the ultimate load capacity of the structure.

Stress analysis alone (Alternative 2.1) would leave considerable uncertainty as to the load capacity of the core support structure.

4. SCHEDULE REQUIRMENTS

Validate the preliminary design by [3/90], six months after the start of the final design phase.

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Design alternative 2.1 would be employed with increased risk that a satisfactory position on the structural integrity and reliability of this component might not be developed in time for the final design.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CONFIRM HOT DUCT INTEGRITY
DDN M.10.17.25
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insulated metallic duct complete with bellows has been selected to transport the primary coolant from the reactor vessel core outlet plenum to the steam generator vessel. In addition to high temperature, high velocity gas, and pressure differentials, the components are subjected to potentially detrimental flow-induced and acoustically induced vibration. The two most susceptible components to the vibration environment are the thermal barrier and the bellows. Very limited data is available on the effects of vibration on components such as these, especially since the effects are configuration-sensitive. Hence, validation testing of an assembly is required.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.1.1.1.1.1 "Maintain Integrity of Hot Duct," Assumption #4: Hot duct components will satisfactorily sustain main circulator output energy levels up to 160 dB and frequencies up to 4200 Hz and plant transients for the life of the plant.

F2.1.2.2.1.1.1.1.1 "Protect the Capability to Maintain Integrity of Hot Duct", Assumption #4: Hot duct components specifically the thermal barrier and bellows will be fully functional for the life of the plant.

1.2 Current Data Base Summary

A series of acoustic tests on flat and curved thermal barrier components was completed in 1984. During 1974-1975 a series of cyclic thermal tests were conducted on a 0.6 (34 in.) scale model of a hot duct at CEA's Chela facility. None of tests are directly applicable to the current 350 MW(t) design.

1.3 Data Needed

Data satisfying Quality Assurance Level II are needed to confirm the response of the thermal barrier and the bellows to the vibration environment in order to give reasonable assurance that the components have integrity for the life of the plant.

1.4 Data Parameters/Service Conditions

a. Service temperatures, ZC (ZF)

Hot side (mixed mean)	687 (1268)
Cold side	258 (497)

b. Operating environment

Primary coolant	Helium
Pressure	6.60 MPa (925 psia)
Velocity	[44-61 m/s (145-200 ft/s)]
Sound pressure level	[160 dB]
Maximum fluence	10^{17} n/cm^2 total neutron fluence
Depressurization	152 kPa/s (22 psi/s)

c. Startup/shutdown cycles [500]

2. DESIGNER'S ALTERNATIVES

2.1 Replace bellows with slip joint and sliding seals joint.

2.2 Reduce the size of thermal barrier coverplates, hence increase the effective damping of the components, thereby decreasing damage potential.

2.3 Complete the design on the basis of data from German and Japanese hot gas duct development programs.

2.4 Design thermal barrier using a nonfibrous insulation material.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The proposed method consists of fabricating and testing a full-size section hot duct model including pipe, bellows, and thermal barrier made out of the selected materials. The selected approach is intended to overcome design and data deficiencies (such as strain and vibration response). It is concluded that the increased confidence in design generated by the full scale model test is cost effective relative to the alternatives given in Section 2. Specifically, the alternatives were rejected for the following reasons: (1) a nonbellows arrangement that would accommodate thermal movements and installation/removal requirements would most likely have significant bypass flow; (2) reducing the size of the coverplates will increase component costs, increase heat shorts, increase risk of component failure (due to additional parts), (3) the German and Japanese hot gas duct designs are sufficiently different from the reference design that very little useful data would be applicable, and (4) a nonfibrous insulation thermal barrier would require an extensive validation test program for verification of adequacy.

4. SCHEDULE REQUIREMENTS

Validate the design by (9/90) at least 3 years before the end of the final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be employed with increased risk that a satisfactory position on the structural integrity and reliability of this component might not be developed in time for the final design.

REV

W E Pack 3-16-87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/19/87

DETERMINE EFFECTS OF IRRADIATION ON PROPERTIES OF ALLOY 800H
DDN M.10.17.26
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insufficient irradiation property data base exists for the design of the hot duct thermal barrier fabricated from Alloy 800H base metal and weldments.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.1.1.1.1.1, Maintain Integrity of Hot Duct (Assumption 2). In service irradiation will not significantly degrade the design properties of metal used in hot duct.

1.2 Current Data Base Summary

Alloy 800 and 800H have extensive tensile and creep rupture data at total fluences above 10^{20} nvt which show that irradiation typically has little effect on tensile strength but reduces ductility and creep rupture strength. These data indicate that if Alloy 800H is irradiated at temperatures below 1100°F, this effect will be acceptable if total fluence does not exceed 10^{20} nvt. At higher temperatures the reduction in ductility becomes more pronounced. Limited data on Hastelloy X irradiated and tested at 1200°F show that if the fluence exceeds 10^{17} nvt thermal plus 10^{17} nvt fast there is a factor of two decrease in ductility. No similar data is available for Alloy 800H.

1.3 Data Needed

Data are needed on the effects of irradiation at temperatures from 1100°F to 1400°F on selected properties of Alloy 800H. Of interest is the effect of such exposure on subsequent service capability of Alloy 800H hot duct. These data need to be sufficient to quantify to the same confidence as the ASME B&PV Code that these properties meet or exceed design values. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Sufficient data are required to determine the effect of irradiation in helium at temperatures from 1100°F to 1400°F and at fluence levels from 10^{18} n/cm^2 thermal (E $\leq$ 2.4 eV) and 10^{18} n/cm^2 epithermal (E > 2.4 eV) to 10^{19} n/cm^2 thermal and 1019 n/cm^2 epithermal on the following properties of Alloy 800H.

a. Tensile properties in helium at temperatures from 400°F to 1600°F with design values for:

- o Minimum ultimate tensile strength (S_u) of not less than 12 ksi.
- o Elastic modulus (E) within $\pm$ 20% of E for as received value.
- o Minimum ultimate total elongation of at least 10%.
- o Yield strength (S_y) of not less than 4.5 ksi.

b. Low cycle fatigue strain range to 500 cycles in helium at temperatures from 400°F to 1400°F with a design value of at least 0.0004 based on hold times up to 600 h.

c. Fracture toughness properties in helium at temperatures from 400°F to 1400°F with design values for:

- o Critical stress intensity factor (K_{IC}) of greater than [55 MPa $\sqrt{m}$ (50 ksi $\sqrt{in.}$)]
- o Fatigue crack growth rate (da/dN) of less than [2.5 x 10^{-3} mm/cycle (1 x 10^{-4} in/cycle)] for $\Delta K \leq$ 33 MPa $\sqrt{m}$ (30 ksi $\sqrt{in.}$) and R (min. to max. load ratio) = 0.0 to 0.75
- o Creep crack growth rate (da/dt) of less than [1.3 x 10^{-9} mm/s (5 x 10^{-11} in./s)] for creep stress $\leq$3 ksi

d. Creep rupture (stress rupture versus time and rupture strain) in helium at temperatures from 1200°F to 1400°F for duration up to 300,000 h with design values for:

- o Minimum rupture stress (S_R) of not less than 3 ksi.
- o Minimum stress to give 1% strain (S_1) of not less than 2 ksi.

e. High cycle fatigue strength to 3x10^{11} cycles in helium at temperatures from 1100°F to 1400°F with a design value of at least [6.0 ksi].

2. DESIGNER'S ALTERNATIVES

The alternatives are as follows:

2.1 Limit lifetime fluence to less than [10^{17} n/cm^2 thermal (E $\leq$ 2.4 eV) and 10^{17} n/cm^2 epithermal].

2.2 Use alternate materials, such as graphite and carbon-carbon composites which are more resistant to damage from irradiation.

2.3 Design conservatively to account for limited data for irradiated Alloy 800H.

2.4 Design hot duct without thermal barrier on inside surface of hot duct.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The proposed solution consists of using Alloy 800H for the hot duct thermal barrier and experimentally generating data needed to quantify the effects of irradiation at design temperatures. Also, this solution requires that lifetime fluence be limited to [approximately 10^{18} n/cm^2 thermal and 10^{18} n/cm^2 epithermal].

Alternative 2.1 would require additional shielding which would require enlarging the reactor vessel at a substantial cost increase. Alternative 2.2 involves uncertainties since these materials have not been used in this new application. This could require its own set of DDNs to resolve which could be even more expensive. Alternative 2.3 would make the core lateral restraint and hot duct more expensive and would increase the risk of licensing delays as one tries to justify conservative factors. Alternative 2.4 is a relatively new concept requiring additional evaluation to verify feasibility.

It is judged that the selected method is the most cost effective means to support the design selection.

4. SCHEDULE REQUIRMENTS

Data is needed by (9/90) at least 3 years before end of final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Additional shielding to reduce irradiation levels (Alternative 2.1) possibly in combination with removing thermal barrier from hot duct (Alternative 2.4). The consequences to the program of nonexecution are estimated to be higher cost.

REL

W. S Betts Jr 16 Mar 87
Originator Date

R F Turner 3/16/87
Department Manager Date

for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/19/87

DETERMINE EFFECTS OF PRIMARY COOLANT CHEMISTRY AND TEMPERATURE ON ALLOY 800H
DDN M.10.17.28
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insufficient property data base exists for the design of the hot duct thermal barrier fabricated from Alloy 800H base metal and weldments.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.1.1.1.1.1, Maintain Integrity of Hot Duct (Assumption 1). Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metal used for hot duct.

1.2 Current Data Base Summary

The primary coolant contains impurities [up to 2 ppmv H_2O, 7 ppmv CO + CO_2, 10 ppmv H_2, 2 ppmv CH_4, and 10 ppmv N_2] which can cause corrosion in the form of oxidation, decarburization and carburization. At the design temperatures of the above components (i.e., 1400°F) carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. Surface oxidation (and concurrent carbon transport) can also affect surface sensitive properties such as fatigue and crack growth. Extensive data is available on the degree of corrosion in HTGR primary coolant helium of Alloy 800H as a function of temperature, duration, and impurity level. However, little data is available on how this degree of corrosion affects selected properties of Alloy 800H.

Data is available on the effects of thermal aging to 30,000 h at temperatures to 1500°F on tensile properties and room temperature CVN impact values.

1.3 Data Needed

Data are needed on the effects of corrosion resulting from exposure to primary coolant (He) with its design impurities and thermal aging at design temperatures on selected properties of Alloy 800H. Of interest is the effect of such exposure on subsequent service capabilities of Alloy 800H in the hot duct thermal barrier. These data need to be sufficient to quantify to the same confidence as the ASME B&PV Code that these properties meet or exceed design values.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Sufficient data are required to determine the effects of corrosion and thermal aging at temperatures from 1000°F to 1400°F for up to 300,000 h on the following properties of Alloy 800H and its weldments.

a. Tensile properties in helium at temperatures from 1000°F to 1600°F with design values for:

- Minimum ultimate tensile strength (S_u) of not less than 12 ksi.
- Elastic modulus (E) within ± 20% of E for as received value.
- Minimum ultimate total elongation of at least 10%.
- Yield strength (S_y) of not less than 4.5 ksi.

b. Low cycle fatigue strain range to 500 cycles in MHTGR helium at temperatures from 400°F to 1400°F with a design value of at least 0.0004 based on hold times up to 600 h.

c. Fracture toughness properties in helium at temperatures from 400°F to 1400°F with design values for:

- Critical stress intensity factor (K_{IC}) of greater than [55 MPa $\sqrt{m}$ (50 ksi $\sqrt{in.}$)]
- Fatigue crack growth rate (da/dN) of less than [2.5 x 10^{-3} mm/cycle (1 x 10^{-4} in./cycle)] for $\Delta K \leq$ 33 MPa $\sqrt{m}$ (30 ksi $\sqrt{in.}$) and R (min. to max. load ratio) = 0.0 to 0.75.
- Creep crack growth rate (da/dt) of less than [1.3 x 10^{-9} mm/s (5 x 10^{-11} in./s)] for creep stress $\leq$3 ksi.

d. Creep properties in helium at temperatures from 1200°F to 1400°F for duration up to 300,000 h with design values for:

- Minimum rupture stress (S_R) of not less than 3 ksi
- Minimum stress to give 1% strain (S_1) of not less than 2 ksi.

e. High cycle fatigue strength to 3 x 10^{11} cycles in helium at temperatures from 1100°F to 1400°F with a design value of at least [6 .0 ksi].

2. DESIGNER'S ALTERNATIVES

The alternatives are as follows:

2.1 Change impurity level allowables in primary coolant to minimize property-degrating corrosion phenomenon.

2.2 Use alternate materials.

2.3 Redesign to reduce stress and accommodate additional allowance for changes in material properties.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The proposed solution consists of using Alloy 800H for the hot duct thermal barrier and experimentally generating data needed to quantify the effects of the primary coolant chemistry and temperature.

Alternative 2.1, with moisture addition, could cause additional problems (oxidation) with graphite components. Alternative 2.2 involves uncertainties which could require DDNs to resolve which could be even more expensive. Alternative 2.3 would make the core lateral restraint and hot duct more expensive and would increase the risk of licensing delays as one tries to justify that the additional allowances are adequate.

It is judged that the selected method is the most cost effective means to support the design selection.

4. SCHEDULE REQUIREMENTS

Data is needed by (9/90) at least 3 years before end of final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Redesign components to reduce stresses (Alternative 2.3) possibly in combination with use of alternate materials (Alternative 2.2). The consequences to the program of nonexecution are estimated to be higher cost.

REV

W. S. Betts Jr. 16 Mar 87
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DATE: 2/27/87

FIBROUS INSULATION MATERIAL PROPERTIES
DDN M.10.17.29
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System: 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The reactor vessel-to-steam generator vessel hot duct employs insulation as a means of protecting the structural integrity of the duct and the primary coolant pressure boundary. The insulation is required to retain its resiliency and physical character during normal and off-normal conditions. Isothermal and temperature differential effects on insulation resiliency is lacking. Hence, confirmatory testing of sample assemblies is required.

1.1 Summary of Function/Title/Assumptions

F.1.1.2.2.1.1.1.1.1 "Maintain Integrity of Hot Duct"
Assumption #5: Insulation resiliency will be maintained for the life of the plant with thermal differentials between 500° and 1400°F.

F.2.1.2.2.1.1.1.1.1 "Protect the Capability to Maintain Integrity of Hot Duct"
Assumption #1: Insulation resiliency will be maintained for the life of the plant.

1.2 Current Data Base Summary

A series of long-term thermal tests have been conducted on a variety of insulation materials at various temperatures and compressions to determine resiliency. Subsequently, representative acoustic vibration tests were conducted and many of the materials were found to be seriously affected. Short-term resiliency tests on high grade insulation blankets were conducted with very promising results.

1.3 Data Needed

Data satisfying Quality Assurance Level II are needed to determine the resiliency of candidate high quality insulation blanket materials in order to be assured that the selected material(s) is capable of lasting for the life of the plant.

1.4 Data Parameters/Service Conditions

a. Service temperatures,

Service temperatures,	°C (°F)
Hot duct-hot side (mixed mean)	687 (1268)
Hot duct-cold side	258 (497)
Maximum surface temperature	760 (1400)

b. Operating environment

Primary coolant	Helium
Purity	[<10 ppm oxidants]
Pressure	6.38 MPa (925 psia)
Velocity (hot duct)	[44-61 m/s (145-200 ft/s)]
Sound pressure level (hot duct)	[160 dB]
Maximum fluence	10^{17} n/cm^2 total neutron fluence

c. Startup/shutdown cycles [500]

2. DESIGNER'S ALTERNATIVES

2.1 Reduce effective acoustic sound pressure levels to that which will not cause significant insulation damage.

2.2 Employ an active cooling system thereby eliminating the need for a passive thermal protection.

2.3 Employ thermal barrier using a nonfibrous insulation material.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is intended to satisfy the data deficiencies needed to design a durable thermal barrier. This data will be used to design the assemblies for the hot duct test (DDN M.10.17.25) and to design the UPTPS.

It is concluded that the confidence gained in the design by performing these tests is cost effective and viable relative to the alternatives given in Section 2. Specifically, the alternatives were rejected for the following reasons: (1) reducing the acoustic sound pressure levels by reducing power will significantly affect plant efficiency. Redesigning the main circulator to produce lower noise levels will involve considerable design effort and subsequent testing of the circulator; (2) employing an active cooling system will increase the difficulty of inspecting the vessels and add a complex system of plumbing. Additionally, the presence of cooling tubes could effectively decrease the local vessel temperature during normal operation and would necessitate incorporation of shielding to reduce the influence of radiation on vessel material ductility; and (3) a nonfibrous insulation thermal barrier would require an extensive test program for verification of adequacy.

4. SCHEDULE REQUIREMENTS

Complete all testing by (9/90) at least 3 years before end of final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 2.1 would be employed which would result in an increase in plant cost. At least a limited amount of material property testing would still be required to qualify and characterize the insulation. Without this testing the integrity and reliability of the insulation would be suspect.

REV

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Manager, Project Operations Date

DATE: 3/16/87

VALIDATE THE PRESSURE DROP FROM COLD DUCT ENTRANCE TO CORE INLET
DDN M.10.17.30
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Confirmation is required that the following limits are met: pressure drop from cold duct entrance to core entrance and flow maldistribution among the 11 inlet channels on the core barrel.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.1.1.1.3.1 "Channel Primary Coolant from Cold Cross Duct Entrance to Core Inlet," Assumption 1: Flow loss coefficients from cold duct entrance to core inlet are valid.

1.2 Current Data Base Summary

No data for the specific geometry exist. Estimates for loss coefficients have been made based on data available in the general literature.

1.3 Data Needed

Loss coefficients from cold duct entrance to core inlet as a function of Reynolds number for each of the 11 channels around the core barrel. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Reynolds number range

Maximum	[TBD]
Minimum	[TBD]

2. DESIGNER'S ALTERNATIVES

2.1 Use the estimates based on the data in the general literature.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to determine the pressure loss coefficients by testing. Estimates based on available data for simple flow geometries are highly uncertain. Also, the test may identify modifications to reduce pressure drop and/or improve flow distribution.

4. SCHEDULE REQUIREMENTS

By the end of preliminary design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Use the estimates based on data in the general literature. Consequences are the need to design the circulator for a higher system pressure drop or risk derating the plant. Also, the design may be less optimized and the pressure drop may be unnecessarily high.

REV

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Originator Date

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DATE: 2/27/87

MULTIAXIAL STRENGTH OF GRAPHITE FOR CORE COMPONENTS
DDN M.10.18.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The failure surface defined by the maximum stress failure theory is a simplified approximation whose uncertainty needs to be quantified and included in the probabilistically based stress criteria which are being developed for showing compliance with the Goal 2 reliability requirements.

1.1 Summary of Function/Title/Assumptions

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity." Assumption 1: The maximum stress failure theory is a reasonable approximation for H-451 graphite under multiaxial state of stress.

1.2 Current Data Base Summary

Biaxial data are available on unirradiated ATJ, ATJ-S, Graphite-G, JTA, PGX, and 2020 graphites. No biaxial or triaxial strength tests have as yet been performed on H-451 graphites.

1.3 Data Needed

Data are needed to determine the reduction in the uniaxial strength of core component graphite due to multiaxial stress conditions. The data are needed for bi- and triaxial tension and tension/compression combinations. The data base must be adequate to show with [95]% confidence that the mean value of the uniaxial strength is not reduced by more than:

[15]% in a biaxial stress field
[20]% in a triaxial stress field

The above statistical data base is needed only for unirradiated graphite at room temperature in air. An additioinal small number of data points are needed on the effects of the service conditions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 120°C - 950°C (248°F-1742°F)

Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

Operating Environment: Helium at 1 - 63 atm pressure

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Estimate the error in the maximum stress failure theory on the basis of existing data.

2.2 Eliminate the need for detailed quantification of the errors in the stress analysis by using stress limits with deterministically selected high safety factors from which the Goal 2 reliabilities can be conservatively estimated.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to establish a "mean value" failure surface such that the error in the simplified maximum stress failure theory can be quantified and the effects on the reliabilities calculated. Alternative 2.1 was rejected because an estimate would be difficult to defend. Alternative 2.2 would require a lower power density and a corresponding increase in core volume and hence cost.

4. SCHEDULE REQUIREMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by [10/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 3
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.1 would be used with the risk of rejection during licensing resulting in either a crash technology program or a belated design change.

REV

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Originator Date

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G.C. Bramblett 3.25.87
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DATE: 2/27/87

FATIGUE DATA FOR GRAPHITE FOR CORE COMPONENTS
DDN M.10.18.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Fatigue analysis is required for the graphite core components. In this analysis, the cumulative effects of varying stress amplitudes must be accounted for.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.4, "Maintain Fuel Element Structural Integrity." Assumption 3: Miner's rule for estimating the cumulative fatigue is applicable to H-451 graphite.

1.2 Current Data Base Summary

Some uniaxial push-pull fatigue tests in air at ambient temperature have been made on axial and radial specimens of H-451 graphite from a single billet. The stress amplitude was held constant during each test. The stress ratio, R (ratio between the minimum stress and the maximum stress during a cycle), varied between -1 and 0, and tests were conducted to a maximum of 10^5 cycles. No tests in which the stress amplitude was changed have been conducted.

1.3 Data Needed

The following fatigue data are required:

a. Fatigue life as a function of stress amplitude for H-451 graphite under constant amplitude cyclic loading for two conditions:

1) Cycling between tension and equal compression (stress ratio of -1).

2) Cycling between tension and zero stress (stress ratio of 0).

The data base must be sufficient to determine the mean value of the fatigue strength within ±[6]% at [95]% confidence.

b. Fatigue life for H-451 graphite subjected to sequential series of cycles with different amplitude. Enough data are needed to establish the difference between constant amplitude fatigue life and varying amplitude fatigue life with [95]% confidence.

The above data are needed for unirradiated graphite at room temperature. In addition, a limited number of data points are needed to determine the effects of the operating environment. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 120°C - 950°C (248°F - 1742°F)

Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

Operating Environment: Helium at 1 - 63 atm pressure

Maximum Number of Load Cycles from Plant Cycles: [10^2 < TBD < 10^5]

Maximum Number of Cycles from Seismic Vibrations: [10^2 < TBD < 10^5]

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Decrease the maximum tensile stress in the graphite to well below the fatigue endurance limit, in which case an adequate fatigue strength is ensured without any fatigue analysis.

2.2 Complete the fatigue analysis on the basis of the existing data base and assume Miner's rule to be valid without validation.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to perform fatigue analysis using Miner's rule and to validate the applicability of this rule to H-451 graphite.

The use of decreased stress (2.1) is judged to be significantly less attractive since it requires lower core power densities, which result in increased capital and operating costs. Alternative 2.2 would incur the risk of rejection during licensing.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by [10/91], two years after the start of the final design phase.

[80-68]

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be used with the risk of rejection during licensing resulting in either a crash technology program or a belated design change.

REV

W. Gorholt 3/16/87
Originator Date

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Department Manager Date

G. C. Bramblett 3.25.87
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DATE: 2/27/87

STATISTICS OF MECHANICAL PROPERTIES OF GRAPHITE CORE COMPONENTS
DDN M.10.18.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Probabilistically based stress criteria are used to ensure compliance with the Goal 2 reliability requirements. The statistical variability of the mechanical properties of the graphite core components is needed for the development of these criteria.

1.1 Summary of Function/Title/Assumptions

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."

Assumption 2: The coefficients of variation given in the Graphite Design Data Manual are accurate to within [25]%.

Assumption 3: Confidence limits on the mean value and standard deviation can be sufficiently determined through the t and Chi-Square distributions, respectively.

Assumption 4: Uncertainty in the estimation of the skewness of a distribution can be estimated by the uncertainty in the coefficient of variation.

1.2 Current Data Base Summary

Tensile, compressive, and flexural strengths in air at ambient temperature have been measured on axial and radial specimens on approximately 100 billets of preproduction and production H-451 graphite from six fabrication lots. For one billet, specimens were taken throughout the whole volume; for 12 billets, specimens were taken from four locations; and for the remainder of the billets specimens were taken from two locations. No strength tests have been made at elevated temperatures. Analysis of the currently available statistical data indicates a strong negative skewness, which is being interpreted as a bimodal normal distribution on the basis of flaw analysis.

A considerable body of data exists on the effects of irradiation on Young's modulus and tensile strength of H-451 graphite specimens irradiated in the Oak Ridge Reactor at 550°C through 1300°C to fluences between 1 x 10^{21} neutrons/cm^2 and 1 x 10^{22} neutrons/cm^2. Specimens were taken from five billets selected from three different production lots. Additional sonic modulus data on H-451 graphite irradiated at 600°C and 900°C to fluences up to 4 x 10^{22} neutrons/cm^2 are available from HFIR capsule irradiations.

1.3 Data Needed

Data are needed to define the tensile and compressive strengths, Poisson's ratio, and stress-strain relationship in accordance with appropriate ASTM standards for H-451 graphites, including the effects of:

a. Orientation and location in billet.

b. Variation from billet to billet and from lot to lot.

c. Temperature ranging from shutdown conditions to the maximum service temperature.

d. Fast neutron fluence.

The data base must be sufficient to establish at [95]% confidence that the mean values of the required properties lie within the following bounds:

Tensile strength:	±[3]%
Compressive strength:	±[10]%
Stress-Strain relationship:	±[8]%
Poisson's Ratio:	±[20]%

In addition to the above statistical data base, information is also needed on the effects of volume and of a pressurized helium environment. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 120°C - 950°C (248°F - 1742°F)

Operating Environment: Helium at 1 - 63 atm pressure

Maximum Fluence: 5 x 10^{25} n/m^2 (E > 29 fJ, HTGR)

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the existing data base.

2.2 Eliminate the need for a detailed statistical data base by using stress limits with deterministically selected high safety factors from which the Goal 2 reliabilities can be conservatively estimated.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to show compliance with the Goal 2 requirements through detailed probabilistic methods. This requires an adequate statistical data base.

Alternative 2.1 was rejected because the existing data base is inadequate for a probabilistic design approach. Alternative 2.2 may require a lower power density and, consequently, a larger core. The effects on capital and operating cost could be large.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by [10/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 could be used. The consequences could be a reduced power density and a corresponding increase in core size resulting in higher capital and operating cost.

REV

W. Gorholt 3/16/87
Originator Date

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DATE: 2/27/87

STATISTICS OF IRRADIATION-INDUCED STRAIN OF GRAPHITE CORE COMPONENTS
DDN M.10.18.04
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Probabilistically based stress criteria are used to ensure compliance with the Goal 2 reliability requirements. The statistical variability of the irradiation induced strain of the core component graphite is needed for the development of these criteria.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.4, "Maintain Fuel Element Structural Integrity."

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."

Assumption 1: The mean values of the irradiation-induced dimensional changes given in the Graphite Design Data Manual for H-451 graphite are valid.

Assumption 2: The coefficients of variation given in the Graphite Design Data Manual are accurate to within [25]%.

Assumption 3: Confidence limits on the mean value and standard deviation can be sufficiently determined through the t and Chi-Square distributions, respectively.

Assumption 4: Uncertainty in the estimation of the skewness of a distribution can be estimated by the uncertainty in the coefficient of variation.

1.2 Current Data Base Summary

A considerable body of data exists on the effects of irradiation on dimensional changes of H-451 graphite specimens irradiated in the Oak Ridge Reactor at 550°C through 1300°C to fluences between 1×10^{21} neutrons/cm^2 and 1×10^{22} neutrons/cm^2. Specimens were taken from five billets selected from three different production lots. Additional dimensional change data for H-451 graphite irradiated at 600°C and 900°C to fluences up to 4×10^{22} neutrons/cm^2 are available from HFIR capsule irradiations.

1.3 Data Needed

Data are needed to define the irradiation-induced dimensional changes of graphite H-451 as a function of fluence and temperature, including:

a. Dependence on orientation and location in billet.

b. Variation from billet to billet and lot to lot.

The data base must be sufficient to determine the mean values of the irradiation strains within $\pm$[0.05]% strain at [95]% confidence. Some additional data are needed to establish the within billet correlation. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 300°C - 950°C (572°F - 1742°F)

Operating Environment: Helium at 1 - 63 atm pressure

Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the existing data base.

2.2 Eliminate the need for a detailed statistical data base by using stress limits with deterministically selected high safety factors from which the Goal 2 reliabilities can be conservatively estimated.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to show compliance with the Goal 2 reliability requirements through detailed probabilistic methods. This requires an adequate statistical data base.

Alternative 2.1 was rejected because the existing data base is inadequate for a probabilistic design approach. Alternative 2.2 may require a lower power density and, consequently, a larger core.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by [10/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be used. The consequences may be a reduced power density and a corresponding increase in core size resulting in higher capital and operating costs.

REV

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DATE: 2/27/87

STATISTICS OF IRRADIATION-INDUCED CREEP OF GRAPHITE CORE COMPONENTS
DDN M.10.18.05
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Probabilistically based design criteria are being used for showing compliance with the Goal 2 reliability requirements. The statistical variability of the creep properties of the core component graphite is needed for the development of these criteria.

1.1 Summary of Function/Title/Assumptions

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."

Assumption 1: The coefficients of variation given in the Graphite Design Data Manual are accurate to within [25]%.

Assumption 2: Confidence limits on the mean value and standard deviation can be sufficiently determined through the t and Chi-Square distributions, respectively.

Assumption 3: Uncertainty in the estimation of the skewness of a distribution can be estimated by the uncertainty in the coefficient of variation.

1.2 Current Data Base Summary

Two compressive creep capsules without continuous strain registration operating at 600°C and three compressive creep capsules operating at 900°C have been completed at ORNL. Each capsule contained eight creep specimens of H-451 graphite stressed to 2000 psi or 3000 psi in compression. Two specimens of H-451 graphite were irradiated in tensile creep assemblies with continuous strain registration at Petten. The temperature was 820°C to 850°C and the stress was 870 psi. The ORNL experiments included measurements for the effect of creep strain on Young's modulus, Poisson's ratio, and thermal expansivity.

1.3 Data Needed

The following data are needed for H-451 graphite as function of fluence and temperature:

a. Steady state creep strain in tension and compression up to 1% creep strain.

b. Transient (primary) creep strain.

c. Transverse-to-longitudinal strain ratios.

The data base must be sufficient to establish the mean value of the steady state creep strain within ±[8]% and the mean values of the other creep properties within ±[20]% both with [95]% confidence. In defining the required creep properties, the following effects need to be included.

a. Dependence on orientation and location in billet.

b. Variation from billet to billet and from lot to lot.

In addition to the statistical data base, some data are also needed to establish the effect of creep on tensile strength, Young's modulus, thermal expansivity and thermal conductivity. Furthermore, data are needed to validate that the creep strain is not significantly affected by the flux level. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 300°C - 950°C (572°F - 1742°F)

Operating Environment: Helium at 1 - 63 atm pressure

Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the existing data base.

2.2 Eliminating the need for a detailed statistical data base by using stress limits with deterministically selected high safety factors from which the Goal 2 reliabilities can be conservatively estimated.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to show compliance with the Goal 2 reliability requirements through detailed probabilistic methods. This requires an adequate statistical data base.

Alternative 2.1 was rejected because the existing data base is inadequate for a probabilistic design approach. Alternative 2.2 may require a lower power density and, consequently, a larger core.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by [9/91], two years after the start of the final design phase.

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be used. The consequences may be a reduced power density and a corresponding increase in core size resulting in higher capital and operating costs.

REV

W. Gotholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

STATISTICS OF THERMAL PROPERTIES OF GRAPHITE FOR CORE COMPONENTS
DDN M.10.18.06
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The thermal properties are needed to complete the thermal-hydraulic design of the graphite core components. The statistical variability of these properties is needed to develop probabilistically based stress criteria for showing compliance with the Goal 2 reliability requirements.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.1, "Transfer Heat from Fuel to Heat Transfer Surface."

Assumption 1: The mean values of thermal expansivity, thermal conductivity, and specific heat given in the Graphite Design Data Manual for H-451 graphite are valid.

Assumption 2: The lower bound for the emissivity of H-451 graphite is 0.8.

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."

Assumption 1: The coefficients of variation given in the Graphite Design Data Manual are accurate to within [25]%.

Assumption 2: Confidence limits on the mean value and standard deviation can be sufficiently determined through the t and Chi-Square distributions, respectively.

Assumption 3: Uncertainty in the estimation of the skewness of a distribution can be estimated by the uncertainty in the coefficient of variation.

1.2 Current Data Base Summary

Axial and radial thermal expansion measurements have been made on specimens from ten billets of production H-451 graphite. In most cases, specimens have been from four locations in the billet. Measurements were made between room temperature and 500°C.

Thermal diffusivity measurements in the axial and radial direction have been made on seven billets of production H-451 graphite between room temperature and 800°C. Specimens were taken from one or two locations in the billet.

Data exists on the effects of irradiation on thermal diffusivity and thermal expansivity of H-451 graphite specimens irradiated in the Oak Ridge Reactor at 550°C through 1300°C to fluences between 1×10^{21} neutrons/cm^2 and 1×10^{22} neutrons/cm^2. Specimens were taken from five billets selected from three different production lots. Additional thermal expansivity data on H-451 graphite and some data on early subsize prototype H-451I graphite irradiated at 600°C and 900°C to fluences up to 4×10^{22} neutrons/cm^2 are available from HFIR capsule irradiations.

1.3 Data Needed

Thermal expansivity, conductivity, emissivity, and specific heat are needed for graphite H-451, including:

a. Dependence on orientation and location in billet.

b. Variation of within and between billets and from lot to lot.

c. Temperature dependence.

d. Dependence on neutron fluence and irradiation temperature.

The data base must be sufficient to establish a [95]% confidence that the mean values of the required thermal properties are within the following bounds:

Expansivity:	±[5]%
Conductivity:	±[5]%
Specific Heat:	±[10]%

For emissivity, [95]% confidence is needed that the lower bound of the property is [0.80]. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 120°C - 950°C (248°F - 1740°F)

Operating Environment: Helium at 1 - 63 atm pressure

Maximum Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the existing data base.

2.2 Eliminate the need for a detailed statistical data base by using stress limits with deterministically selected high safety factors from which the Goal 2 reliabilities can be conservatively estimated.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to show compliance with the Goal 2 reliability requirements through detailed probabilistic methods. Alternative 2.1 was rejected because the existing data base is inadequate for a probabilistic design approach. Alternative 2.2 may require a lower power density and, consequently, a larger core.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by 10/91, two years after the start of the final design phase.

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be used. The consequences may be a reduced power density and a corresponding increase in core size resulting in higher capital and operating costs.

REV

W. Gorholt 3/16/87
Originator Date

R7 Turner 3/16/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

STATISTICS OF FRACTURE MECHANICS PROPERTIES OF GRAPHITE FOR CORE COMPONENTS
DDN M.10.18.07
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

To meet the Goal 2 reliability requirements for the core components, it is necessary to calculate the probability of functional damage. Functional damage has been defined as a crack extending all the way across a fuel or reflector element or at least a significant distance into the element. So far, only vertical cracks have been addressed using existing continuum mechanics methods. Fracture mechanics methods are needed to address horizontal cracks and also to validate the continuum mechanics methods.

1.1 Summary of Function/Title/Assumptions

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."

Assumption 3: The progression of vertical cracks (due to radial stresses) can be analyzed with continuum mechanics methods.

Assumption 4: Horizontal cracks which need fracture mechanics methods are less probable than vertical cracks due to (1) lower stresses in the axial direction, and (2) only vertical cracks have been observed in FSV fuel elements.

1.2 Current Data Base Summary

Some static K_{IC} measurements have been made on production H-451 graphite using the chevron-notched short-rod specimen geometry. A few measurements have also been made on specimens from early subsize prototype H-451I billets. Changes in the static K_{IC} have been measured on H-451 graphite specimens irradiated at 600°C and 900°C to fluences of 1.6 $\times 10^{22}$ n/cm^2 in HFIR.

1.3 Data Needed

A data base is needed to define the critical stress intensity factors (K_{IC}) and strain energy release rates (G_{IC}) for crack initiation, stable crack growth, and crack arrest for graphite H-451 at room temperature in air, including:

a. The effects of orientation and location in billet.

b. Variation from billet to billet and from lot to lot.

The data base must be sufficient to establish the mean values of the above fracture mechanics properties within ±[10]% at [95]% confidence.

A limited number of additional data points are needed to establish the effects of the operating environment on the fracture mechanics properties. The environmental conditions whose effects need to be established are:

a. The effect of irradiation.

b. The effect of temperature within the service temperature range.

c. The effect of pressurized helium.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Service Temperature Range: 300°C - 950°C (572°F - 1740°F)

Operating Environment: Helium at 1 - 63 atm pressure

Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the existing finite element codes to examine cracking initiation and progression based upon the maximum principal stress failure theory for vertical and also for crack initiation for horizontal cracks.

2.2 Eliminate the need for fracture mechanics methods by designing the core graphite components with high safety factors from which the Goal 2 reliabilities can be conservatively estimated.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to develop fracture mechanics methods and use these to study crack propagation as part of the analysis for showing compliance with Goal 2 reliability requirements. Alternative 2.1 was rejected because it is theoretically unsound and thus would have been difficult to defend to the NRC. Alternative 2.2 may require a lower power density and, consequently a larger core. The effects on capital and operating cost would be large.

4. SCHEDULE REQUIRMENTS

Preliminary data are needed by [10/88] one year after the start of preliminary design and final data by 10/91, two years after the start of the final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be used. The consequences may be a reduced power density and a corresponding increase in core size resulting in higher capital and operating cost.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

CORROSION CHARACTERISTICS OF CORE COMPONENTS GRAPHITE
DDN M.10.18.08
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite core components may be corroded by coolant impurities, principally H_2O, with consequent deterioration of their integrity. Associated data needs are DDNs: M.10.01, M.10.18.09.

1.1 Summary of Function Number/Title/Assumptions

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity." Assumption 4: The existing correlations for H-451 graphite corrosion are accurate within a factor of 2 at 95% confidence.

F3.1.1.2.1.1.2.1.1.3 "Maintain Controllable Geometry." Assumption: The correlations for H-451 graphite are accurate within a factor of 2 at 95% confidence.

1.2 Current Data Base Summary

The corrosion of core component graphite (H451) by coolant impurities (H_2O, O_2, and CO_2) may be mass-transfer limited, chemical-reaction limited, or a combination of both; consequently, both processes must be characterized. The transport of coolant impurities is a combination of pore diffusion and permeation flow due to pressure gradients; the transport rates increase with increasing graphite burnoff. The former process is characterized by an effective diffusion coefficient; the reference correlation was obtained for H_2O transport in H451 graphite with 1% burnoff. The permeability of H451 has not been well characterized.

The reference correlations for the kinetics of H451 corrosion by coolant impurities are based primarily upon laboratory measurements on small unirradiated specimens in helium with high impurity levels at or near atmospheric pressure. Some data were obtained at elevated pressures (~20 atm) in the High Pressure Test Loop. Since the measurements were all made on unirradiated graphite, the effects of radiolysis and catalysis by fission metals on the graphite corrosion rate were not systematically investigated. The reaction of H_2O with H451 exhibits Langmuir Hinshelwood type kinetics with significant product inhibition by H_2 but not by CO. The reference correlation for oxidation of H451 by air was derived from lab measurements on H327 graphite.

1.3 Data Needed

Correlations describing the corrosion of H451 graphite by coolant impurities during normal operation and H_2O ingress events are needed. Data are needed to characterize both the transport of coolant impurities and graphite corrosion products in H451 graphite and the intrinsic kinetics for the reaction of water and oxygen with H451 graphite. To characterize the transport of coolant impurities in graphite, the porosity, tortuosity, and permeability of the graphite must be determined. To characterize the reaction kinetics, the reaction rate must be determined as a function of temperature, impurity concentrations, system pressure, and time. In addition, the effects of radiolysis and catalysis by graphite impurities and by fission metals on the reaction kinetics must be determined. Finally, the effects of partial graphite burnoff on both the mass transfer processes and the intrinsic reaction kinetics must be quantified.

Sufficient data are needed to predict the burnoff within an accuracy of [2] with [95]% confidence. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

a. Normal Operation

Environment	Helium
Maximum fast fluence (E >29 fJ, HTGR)	[5 x 10^{21}] n/cm^2
Maximum gamma flux	[TBD] MeV/cm^2-s
Primary coolant temperature range	[120 to 700]°C
Graphite temperature range	[120 - 950]°C
Maximum time averaged coolant impurity levels	[2] ppm H_2O [5] ppm CO [2] ppm CO_2 [TBD] ppm O_2 Total Oxidants <10 ppm [10] ppm H_2
Helium coolant pressure	1 to 63 atms

b. Moisture Ingress Conditions

	Maximum Concentration (ppmv)
Moisture ingress with steam generator dump failure (DBE-9)	660
Moisture ingress with moisture monitor failure (DBE-8)	18,000

	Amount of Water Leaking into Reactor Vessel
Moisture ingress without steam generator dump (SRDC-6, 7)	1820 lb

c. Air Ingress Conditions

	Amount of Air Ingress
Depressurized conduction cooldown (SRDC-10	21 lb-mole

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the current data base and add more margin to account for the uncertainties.

2.2 Impose tighter tech specifications on primary coolant oxidant levels.

2.3 Use a higher purity, more corrosion resistant graphite.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to determine the corrosion characteristics of H451 graphite under normal operating and H_2O ingress conditions. Design alternative 2.1 is rejected because the uncertainties in the current data base would necessitate unacceptably large design margins. Design alternative 2.2 is rejected because imposition of tighter tech spec limits on coolant impurities is expected to adversely impact plant availability. Design alternative 2.3 is rejected because development and qualification of a higher purity graphite would add significant development costs. (H-451 is already a graphite with high oxidation resistance.)

4. SCHEDULE REQUIREMENTS

Preliminary data by [3/89], 6 months prior to PSSAR submittal and final data by [10/91], two years after the start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

A combination of Alternatives 2.1 and 2.2 with the necessity of added margins in the design to compensate for uncertainties in the extent of core component corrosion. The consequence would be unnecessarily restrictive tech specs on primary coolant impurity levels which could have an adverse impact on plant availability.

REV

W. Gotholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

for G. C. Bramblett 3.25.87
Manager, Project Operations Date

[80-88]

DATE: 2/27/87

CORROSION EFFECTS ON CORE COMPONENT GRAPHITE DESIGN PROPERTIES
DDN M.10.18.09
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite core components must withstand the loads imposed on them without exceeding the stress limits of Goal 1 and the reliability limits of Goals 2 and 3 including degradation due to corrosion effects. Thus, the relationship between the design properties and the amount and distribution of oxidation is needed.

Associated data needs are DDNs M.10.01 and M.10.18.08.

1.1 Summary of Function/Title/Assumptions

F2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."

Assumption 5: The existing data base is sufficient to predict the corrosion effects on H-451 graphite within a factor of 2 at 95% confidence.

F1.1.2.1.2.2.4, "Maintain Fuel Element Structural Integrity."

Assumption 8: The existing data base is sufficient to predict the corrosion effects on H-451 graphite within a factor of 2 at 95% confidence.

F3.1.1.2.1.1.2.1.1.3 "Maintain Controllable Geometry."

Assumption: Same as Assumption 8 above.

1.2 Current Data Base Summary

Changes in the tensile strength and Young's modulus of H-451 graphite uniformly oxidized up to 20% burnoff at 800°C and 1000°C have been reported. The effect of steam oxidation at 900°C and 1000°C on the static elastic fracture toughness has been measured for H-451 graphite. No thermal properties of H-451 have been determined for oxidized material. The measurements have all been made on unirradiated graphite.

1.3 Data Needed

The data base shall be sufficient to define the following for H-451 graphite:

a. Effects of uniform burnoff on Young's modulus, tensile strength, compressive strength, irradiation-induced creep and dimensional change, fracture toughness, thermal conductivity, thermal expansivity, and specific heat.

b. The effects of nonuniform burnoff on these mechanical and thermal properties.

Data are needed to predict the degradation of the design properties due to graphite corrosion within an accuracy of [2] with [95]% confidence. Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The reactor operating or service conditions are given below.

a. Normal Operation

Maximum fast fluence (E > 29 fJ)	[5.0] x 10^{21} n/cm^2
Primary coolant temperature range	[120°C to 700°C]
Core component graphite temperature range	[120°C to 950°C]
Maximum time averaged coolant impurity levels	[2] ppm H_2O [5] ppm CO [2] ppm CO_2 Total oxidants <10 ppm
Helium coolant pressure	1 to 63 atms

b. Moisture Ingress Conditions

	Maximum Concentration (ppmv)
Moisture ingress with steam generator dump failure (DBE-9)	660
Moisture ingress with moisture monitor failure (DBE-8)	18,000

	Amount of Water Leaking into Reactor Vessel
Moisture ingress without steam generator dump (SRDC-6, 7)	1820 lb
c. Air Ingress Condition	Amount of Air Ingress
Depressurized conduction cooldown (SRDC-10)	21 lb-mole

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Use the current data base in the thermal and stress analysis and add more margin to account for uncertainties.

2.2 Impose tighter tech specifications on the primary coolant oxidant levels.

2.3 Use a higher purity, more corrosion resistant graphite.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to obtain an improved materials data base on the effects of corrosion of the core component graphite. Design alternative 2.1 is rejected because the uncertainties in the current data base would necessitate unacceptably large design margins. Design alternative 2.2 is rejected because imposition of tighter tech spec limits on coolant impurities is expected to adversely impact plant availability. Design alternative 2.3 is rejected because development and qualification of a higher purity graphite would add significant development costs. (H-451 is already a graphite with high corrosion resistance).

4. SCHEDULE REQUIREMENTS

Preliminary data by [3/89], six months prior to PSAR submittal, and final data by [10/91] two years after the start of the final design.

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

A combination of alternatives 2.1 and 2.2 with the necessity of added margins in the design to compensate for uncertainties in the degradation of the core components due to corrosion. The consequence would be unnecessarily restrictive tech specs on primary coolant impurity levels which could have an adverse impact on plant availability.

REV

W. Gotholt 3/16/87

Originator Date

R F Turner 3/16/87

Department Manager Date

for

G. C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

DESTRUCTIVE/NDE DATA FOR CORE GRAPHITE SPECIFICATIONS
DDN M.10.18.10
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Nondestructive testing techniques are needed for product control during procurement of graphite for the core components.

Associated data needs are DDNs: M.10.18.01 and M.10.18.03.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity."
Assumption 2: The production graphite will have the same properties as the design data base.

1.2 Current Data Base Summary

ORNL has developed nondestructive testing techniques and produced data on their accuracy and limitations for graphite grade H-451. The data base is too small to sufficiently validate these techniques for use in product control of mass produced graphite.

1.3 Data Needed

Data are needed to validate NDE techniques and write material control specifications for the procurement of graphite for core components. The NDE techniques must be sufficiently accurate to 1) detect unacceptable flaws in the billets, and 2) determine the tensile strength of smaller specimens with an error no greater than [10]%. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

NDE will be conducted at room temperature conditions.

2. DESIGNER'S ALTERNATIVES

These alternatives were considered:

2.1 Control the strength of the production material by conventional strength testing only. This would require several tensile test coupons from each billet in addition to more extensive mapping of at least one billet from each lot.

2.2 Include additional design margin to account for uncertainties in the production material.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to use nondestructive examination in combination with a minimum of destructive testing to ensure that the production graphite satisfies the specifications. This is the most cost effective approach since NDE is less expensive than conventional destructive testing (Alternative 2.1). Alternative 2.2 could lead to a reduction in allowable stresses and hence a reduced power density and a larger core with a large increase in both capital and operating costs.

4. SCHEDULE REQUIREMENTS

The data is needed by [9/92], before starting procurement of the graphite for the core components. This is assumed to be one year before the end of the final design phase [9/93].

5. PRIORITY

Urgency: 5
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternatives 2.1 would be used which may result in higher cost of the graphite material.

REV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

[illegible] for
G. E. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

VALIDATE FUEL ELEMENT DYNAMIC STRENGTH PREDICTIONS
DDN M.10.18.11
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite core components (fuel and replaceable reflector elements) must withstand dynamic stresses due to seismic loads in combination with thermal/irradiation induced stresses.

1.1 Summary of Function/Title/Assumption

F1.1.2.1.2.2.4, "Maintain Fuel Element Structural Integrity."

Assumption 9: The dynamic strength can be predicted with static finite element methods.

Assumption 10: Thermal/irradiation stresses and seismic stresses can be linearly combined.

1.2 Current Data Base Summary

To address the issue in Assumption 9, a series of dynamic tests of unirradiated FSV fuel elements was performed in 1976 in a pendulum rig. Those tests indicated that the dynamic strength can reasonably well be predicted with static finite element methods. However, all the test specimens were control fuel elements which are not used in the 350 MW(t) core and most of them were made of graphite grade H-327. Only two specimens were from H-451 which is the 350 MW(t) reference material. No data exist on the mechanical strength (static or dynamic) of irradiated fuel elements.

1.3 Data Needed

The failure load of H-451 fuel elements subjected to dynamically applied forces are needed. The nature of the forces and their duration must be representative for the type of loads imposed on the fuel elements during earthquakes. Data points are needed for both virgin fuel elements and irradiated fuel elements with residual stresses resulting from long time exposure to temperatures and fluences comparable to the conditions in an HTGR core.

The data base must be sufficient to establish with [95%] confidence that the analytical methods are conservative, i.e., the mean values of the experimentally determined failure loads are higher than the corresponding analytical predictions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Material: Graphite Grade H-451
Service Temperature Range: 120°C-950°C (248°F-1742°F)
Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)
Operating Environment: Helium at 1-63 atm pressure

2. DESIGNER'S ALTERNATIVES

These alternatives were considered:

2.1 Redesign the core to eliminate the seismic impact loads on the fuel and reflector elements. This would require keying or clamping of the core.

2.2 Proceed on the basis of the present assumptions (static methods, linear combination) without validation.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to validate the dynamic strength predicted on the basis of static methods and linear load combinations. This was seen as the most cost effective approach. Alternative 2.1 would result in a costlier design and more complicated refueling. Alternative 2.2 would incur the risk of rejection during licensing.

4. SCHEDULE REQUIREMENTS

Data are required by [3/90], six months after the start of final design.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design Alternative 2.2 would be used with the risk of rejection during licensing which may result in either a crash technology program or a belated design change.

KEV

W. Gorholt 3/16/87
Originator Date

R F Turner 3/16/87
Department Manager Date

~~G. E. Bramblett~~ for 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

VALIDATE FUEL ELEMENT FAILURE MODE PREDICTIONS
DDN M.10.18.12
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The steps in the design process for showing compliance with the Goal 2 reliability requirements includes calculations of how a crack, if initiated, would progress until the fuel element is functionally damaged. The methods for performing these calculations need validation.

1.1 Summary of Function/Title/Assumption

F.2.1.2.1.2.4, "Protect the Capability to Maintain Fuel Element Structural Integrity."
Assumption 6: The failure mode, i.e., the crack progression can be predicted with the TWOD finite element code.

1.2 Current Data Base Summary

The failure load and failure mode of virgin fuel elements under mechanical loads were determined in two test programs: first for FSV elements in 1976, then for the 2240 MW(t) elements in 1983. Some analytical correlations were performed, but these did not include crack progression analyses. Limited cracking under thermal/irradiation stresses have been observed in two FSV fuel elements, and reasonably good analytical correlations were achieved. The cracking was, however, far from extensive enough to represent failure in a functional sense. No data exist for failure under combined thermal/irradiation and mechanical loads.

1.3 Data Needed

The failure loads and failure modes are needed for fuel elements subjected to the combination of mechanical and thermal/irradiation loads. The thermal/irradiation stresses must be comparable to those developed in a typical fuel element at shutdown conditions (due to creep there are residual stresses at shutdown of the same magnitude as the operating stresses).

The specific data needed are:

a. Mechanical load at crack initiation
b. Location of crack initiation
c. Mechanical load at ultimate failure
d. Crack path from initiation to ultimate failure

The data base must establish with [90%] confidence that the analytical methods are conservative, i.e., the mean values of the experimental data is higher than the corresponding analytical predictions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Material: H-451
Service Temperature Range: 120°C - 950°C (248°F - 1742°F)
Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

Operating Environment: Helium at 1-63 atm pressure

2. DESIGNER'S ALTERNATIVES

These alternatives were considered:

2.1 Validate the analytical predictions on the basis of the existing data.

2.2 Design the core components with a large margin from where the reliabilities can be conservatively estimated without detailed predictions of failure loads and failure modes.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to perform cracking analysis with the TWOD code as a part of the design process for showing compliance with the Goal 2 reliability requirements and to validate the analytical methods by correlation with an expanded data base. Alternative 2.1 would incur the risk of rejection during licensing. Alternative 2.2 would lead to an uneconomical design.

4. SCHEDULE REQUIREMENTS

The data are needed by [10/89], at start of the final design phase.

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design Alternative 2.1 would be used with the risk that the validation would be deemed unacceptable resulting in either a crash technology program or a belated design change.

REV

W. Gorholt 3/16/87

Originator Date

R F Turner 3/16/87

Department Manager Date

[illegible] for G. C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

VALIDATE CORE COMPONENT SEISMIC LOAD PREDICTIONS
DDN M.10.18.13
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. <u>REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING</u>

The seismic loads imposed on the graphite core components are predicted with the MCOCO computer code which uses the excitation at the core boundary as input. These load predictions must be validated.

1.1 <u>Summary of Function/Title/Assumptions</u>

<u>F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity."</u>
<u>Assumption 11:</u> The seismic loads are correctly predicted by the MCOCO computer code.

1.2 <u>Current Data Base Summary</u>

Tests were run in 1974 on a scaled model of a large HTGR core. These tests validated the MCOCO code in general, i.e., the frequency characteristics and the overall core response. A satisfactory validation of the load predictions was not achieved, however, partly due to instrumentation limitations, partly due to an unresolved question about the scaling laws.

1.3 <u>Data Needed</u>

Data are needed on the structural integrity of the fuel elements, i.e., whether the elements break or are otherwise structurally damaged after having been subjected to a simulated earthquake of an intensity for which the analytical methods would predict damage. Data are also needed on the relative impact velocities between the pairs of elements experiencing the highest impact loads.

The data base must be sufficient to establish a [90%] confidence that the analytical methods are conservative; i.e., the analytically predicted loads are higher than the actual loads. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 <u>Data Parameters/Service Conditions</u>

Material: Graphite Grade H-451
Service Temperature Range: 120°C-950°C (248°F-1742°)
Maximum Fast Fluence: 5×10^{25} n/m^2 (E > 29 fJ, HTGR)

Operating Environment: Helium at 1-63 atm pressure
Seismic Excitation at Core Boundary: [0.5 g OBE]
[0.75 g SSE]

2. DESIGNER'S ALTERNATIVES

These alternatives were considered:

2.1 Redesign the core to eliminate the seismic impact loads on the fuel and reflector elements. This would require keying or clamping of the core.

2.2 Continue to use the existing methods without further validation.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to supplement the data base to complete the validation of the seismic design. Alternative 2.1 would result in a costlier design and more complicated refueling. Alternative 2.2 would incur the risk of rejection during licensing.

4. SCHEDULE REQUIREMENTS

The data are needed by [9/92], one year before FSSAR submittal.

5. PRIORITY

Urgency: 4
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Design alternative 2.2 would be used with the risk of rejection during licensing resulting in a crash technology program and possibly also schedule delays.

REV

W. Goholt 3/16/87
Originator Date

R7 Turner 3/16/87
Department Manager Date

Decision for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

VALIDATE CONTROL ROD SHOCK ABSORBER CHARACTERISTICS
DDN M.10.18.14
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Crushable graphite shock absorbers are installed at the bottom of the control rod channel to protect the graphite core support structure in the event of an accidentally dropped control rod.

1.1 Summary of Function/Title/Assumptions

F2.1.2.1.2.2.2.2.2.2.4 "Absorb Energy from Dropped Control Rod"
Assumption 1: A crushable graphite insert can absorb sufficient energy to protect the core support structure.

1.2 Current Data Base Summary

No data is available on the shock absorption characteristics of graphite in crushable form (e.g., perforated, honeycombed, etc.).

1.3 Data Needed

Data are needed to establish the absorption characteristics for three different variants, as specified by the designer, of a crushable graphite shock absorber. (The absorption characteristic is the energy absorbed in crushing action, expressed as a percentage of the total kinetic energy in the falling body). For each of the three variants, a sufficient data base must be established to provide 90% confidence that the mean value of the data base is at most [20%] different from the true mean value. (Provided the data confirms that all three variants are adequate, the designer will select the most cost effective of the variants). Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Material: Graphite Grade H-451
Service Temperature Range: 21°C-700°C (70°F-1300°F)
Maximum Fast Fluence: [1 x 10^{25}] n/m^2 (E > 29 fJ, HTGR)
Operating Environment: Helium at 1-65 atm pressure
Characteristics of Dropped Control Rod:
    Weight of Rod: [180] lb
    Max Drop Height: [32] ft

2. DESIGNER'S ALTERNATIVES

These alternatives were considered:

2.1 Improve the reliability of the control rod such that the unscheduled outage due to a dropped rod meets the allocations even without any shock absorbing features.

2.2 Use a FSV type metallic shock absorber connected to the lower end of the control rod.

2.3 Eliminate the shock absorber and accept the unscheduled outage resulting from a dropped rod.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to use shock absorbers of crushable graphite with validated characteristics. Validation is necessary because of the uncertainties in making analytical predictions. This approach was found to be more cost effective than the alternatives. Alternative 2.1 would increase the cost of the control rods and would also introduce the difficulties and expenses of proving that their reliabilities were adequate. Alternative 2.2 would require periodic replacements of the shock absorber due to embrittlement of the thin-walled metal structure. Irradiation testing would also be necessary to determine the degradation of the shock absorption characteristics as function of the fast fluence. Alternative 2.3 would require an increase in the unscheduled outage allocation due to the extended shutdown for replacing potentially damaged core support components.

4. SCHEDULE REQUIREMENTS

The data are needed by [9/92], one year before FSSAR submittal.

5. PRIORITY

Urgency: 4
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative 2.2 would be used. This would increase operating costs and also technology cost due to the need for irradiation testing.

REV

W. Gerhardt 3/16/87

Originator Date

R F Turner 3/16/87

Department Manager Date

for

G.C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 3/16/87

VALIDATE CONTROL CHANNEL FLOW PREDICTIONS
DDN M.10.18.15
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Thermal/hydraulic input parameters for computer codes are needed to calculate control channel flow to meet the requirements for control rod temperatures, core exit hot and cold streaks, fuel temperatures, and control block stresses.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.1.3 "Control Flow in Control Rod Channel," Assumption 1: Control rod channel entrance and exit loss coefficients are valid.

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity," Assumption 13: Control rod channel entrance and exit loss coefficients are valid.

F1.1.1.2.2.1.2 "Maintain Integrity of Control Rods," Assumption 3: Control rod channel geometry provides [2%] of the circulator flow rate in these channels.

1.2 Current Data Base Summary

Loss factors at the entrance and exit of the channels are available from tests performed at Commissariat A L'Energie Atomique (CEA) and at GA, but the designs and the control rod channel flow requirements for the MHTGR are different from the reactor designs which were modeled in these tests.

1.3 Data Needed

Flow loss coefficients need to be developed for representative graphite channels with and without the control rod in place. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Reynolds number range

Minimum	[0]
Maximum	[100,000]

2. DESIGNER'S ALTERNATIVES

   2.1 Continue to rely on current data base and use larger than necessary performance margins.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to extend the current data base. This will reduce the control rod channel flow requirement necessary to ensure acceptable control rod temperatures. Lowering the control rod channel flow requirement increases the fuel element coolant channel flow, thus reducing fuel temperatures. Lower control rod channel flow also reduces control block stresses and the potential for flow induced vibrations. (See DDN M.10.18.17.)

4. SCHEDULE REQUIRMENTS

Data are needed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Rely on current data base and revert to larger than necessary performance margins. Consequences of nonexecution are greater potential for control rod flow induced vibrations which must be considered in the control rod design, and increased stresses in the control rod blocks and temperatures in the fuel blocks.

REV

M. C. Henderson 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/16/87

VALIDATE FUEL ELEMENT CHANNEL FLOW PREDICTIONS
DDN M.10.18.16
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Thermal/hydraulic input parameters for computer codes are needed to calculate pressure drops, flow distribution, and heat transfer for the fuel element coolant channels.

1.1 Summary of Function/Title/Assumptions

F1.2.2.1.1 "Transfer Heat from Fuel to Heat Transfer Surface." Assumption 4: Coolant channel friction factor and heat transfer correlations are valid.

1.2 Current Data Base Summary

Heat transfer correlations are currently based on smooth tube data obtained from the literature. The greatest uncertainty in these data are in the transitional flow regime.

Friction factor correlations currently used are based on tests performed at Commissariat A L'Energie Atomique (CEA) and at GA. These tests were performed on a different grade of graphite and a larger coolant channel. In addition, no tests were performed in the transition flow regime.

1.3 Data Needed

Friction factor and Nusselt number correlations need to be developed for representative drilled graphite coolant channels. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Reynolds number range

Minimum	[0]
Maximum	[65,000]

b. Coolant channel flow, kg/s (lbm/s)

Minimum	[TBD (TBD)]
Maximum	[0.023 (0.05)]

c. Operating environment

Primary Coolant	Helium

2. DESIGNER'S ALTERNATIVES

2.1 Continue to rely on the current data base and use larger than necessary performance margins.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to extend the current data base. Better knowledge of the coolant channel friction factor will reduce the uncertainty in the core pressure drop. Better knowledge of the heat transfer coefficient will reduce calculated fuel temperatures. In particular, better data in the transitional flow regime will improve predictions of laminar flow instabilities which can result in fuel damage at low power operation. With this information the flow requirements at low power can be minimized.

4. SCHEDULE REQUIRMENTS

Data will be required by one year after the start of final design (10/90) for use in final design calculations.

5. PRIORITY

Urgency: 3
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Use larger than necessary performance margins. Consequences are higher fuel temperatures and graphite stresses, and higher flow requirements during refueling, shutdown, and startup operation. These will have to be considered in the design.

REV

M. C. Henderson 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

D. Allison for
G. E. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/16/87

VALIDATE CONTROL ROD VIBRATION PREDICTIONS
DDN M.10.18.17
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Confirmation is required that flow-induced vibrations of the control rods do not affect the integrity of the control rods.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.1.2 "Maintain Integrity of Control Rods," Assumption 3: Flow-induced vibrations do not contribute significantly to control rod stresses.

1.2 Current Data Base Summary

Control rod vibration tests were performed in 1975 on the Fort St. Vrain control rod design. These tests showed that the control rods were susceptable to flow-induced vibration. These tests were of a limited nature and did not include the effect of crossflow. No data are available for the longer control rods in the 10-block high core.

1.3 Data Needed

Data are required to confirm that the control rods have no significant flow induced-vibrations under any reactor operating conditions with and without crossflow. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Coolant velocity, m/s (ft/s)

Minimum	[0 (0)]
Maximum	[75 (165)]

b. Coolant density, kg/m^3 (lbm/ft^3)

Minimum	[0.75 (0.01)]
Maximum	[28 (0.36)]

c. Contral channel coolant flow, kg/s (lbm/s)

Minimum	[0 (0)]
Maximum	[82 (180)]

2. DESIGNER'S ALTERNATIVES

2.1 Use analyses and current data base.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to validate the design based on experimental data.
Validation by analysis alone (2.2) is not believed to be sufficiently accurate to provide necessary confidence that vibrations will not occur.

4. SCHEDULE REQUIRMENTS

Data are needed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to rely on analysis and the current data base. The consequence of nonexecution could be that it would be necessary to redesign the control rods, if excessive vibration was encountered at reactor startup. This would result in schedule delays and cost increases.

REV

M. C. Hemler 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

[53-7]

DATE: 3/16/87

VALIDATE CORE CROSSFLOW PREDICTIONS
DDN M.10.18.18
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Data are needed to characterize the flow leakage at fuel and reflector elements interfaces (crossflow) to validate the predictions of coolant and control rod channel flows.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity," Assumption 1: Crossflow does not create unacceptable fuel element stresses.

1.2 Current Data Base Summary

Crossflow tests were performed on flat faced elements (eight row block) at Commissariat a L'Energie Atomique (CEA). Useful data were obtained only for basically one gap. In addition, these tests were not performed with differences in coolant channel pressures, thus crossflow from coolant channel to coolant channel was not measured. Also control rod channel blocks were not tested, so that crossflow to control rod channels is unknown.

Tests were performed at GA Technologies on single interfaces, but for flanged fuel elements (with end seals). Flat faced elements have been chosen for the 350 MW(t) modular reactor.

1.3 Data Needed

Data are needed for loss coefficients as a function of expected crossflow pressure differentials, crossflow gaps, and coolant and bypass gap Reynolds numbers. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Crossflow gap widths, m (in.)

Minimum	[0 (0)]
Maximum	[0.0051 (0.2)]

b. Channel/gap pressure differential, atm (psi)

Minimum	[0 (0)]
Maximum	[0.028 (0.40)]

c. Gap Reynolds number

Minimum	[100]
Maximum	[10,000]

d. Coolant channel Reynolds number

Minimum	[100]
Maximum	[65,000]

e. Coolant channels pressure differential, atm (psi)

Minimum	[0 (0)]
Maximum	[0.345 (5.0)]

2. DESIGNER'S ALTERNATIVES

2.1 Allow for greater uncertainties in the stress and fuel failure limits and design control rods to be stable even with greater crossflow.

2.2 Employ more sophisticated thermal/hydraulic analyses to reduce the uncertainties.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to extend the present data base. To allow greater uncertainties may be impractical because of a small margin on stress safety factors. More sophisticated flow analyses could perhaps reduce uncertainties, but not to the degree that can be achieved with experimental data.

4. SCHEDULE REQUIRMENTS

Results are required by the end of preliminary design (9/89) to permit detailed thermal hydraulic design calculation for final design.

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Allow for greater uncertainties in the stress and fuel failure limits and design the control rods to be stable with higher crossflow. Consequences are difficulty in meeting the stress limits, an increase in postulated fuel failures and difficulty in meeting fission product release limits, and possible difficulties and cost in developing a control rod design which can withstand the higher crossflows. These must be addressed in the design.

REV

M.C. Hen[illegible] 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/24/86

VALIDATE THE ABSENCE OF CORE FLUCTUATIONS
DDN M.10.18.19
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Validate that the 10-block high active core and associated reflectors are stable to mechanical/thermal movements which would cause fluctuations in the outlet coolant temperatures.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.1 "Control Core Bypass Flow," Assumption 1: Absence of temperature fluctuations in core exit gas temperatures will be validated.

1.2 Current Data Base Summary

Analyses and tests were performed to resolve the fluctuations experienced on the Fort St. Vrain reactor. The result of these efforts were the installation of region constraint devices.

A long range testing program was initiated in FY-81 on the large (2240) cylindrical prismatic core. Fluctuation testing was performed on a 1/14-scale, three-dimensional model, a 1/7-scale, two-dimensional model, and a 1/4-scale, single column, three-dimensional model.

1.3 Data Needed

Data are needed to validate that the core array will be stable under the expected heating, flow, and pressure drop conditions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Service temperature range, °C (°F)

Minimum	[120 (248)]
Maximum	[900 (1652)]

b. Operating pressure, atm (psi)

Minimum	[1.0 (14.7)]
Maximum	[65 (955)]

c. Primary coolant flow, kg/s (lbm/s)

Minimum	[0 (0)]
Maximum	[159 (350)]

d. Reactor power, MW

Minimum	0
Maximum	350

e. Operating environment

Primary coolant	Helium

2. DESIGNER'S ALTERNATIVES

2.1 Rely on the current data base and analytical predictions.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to validate the final design by experimentally demonstrating that the design has no potential for fluctuations. There is little confidence that validation can be obtained by analytical prediction.

4. SCHEDULE REQUIRMENTS

Data are required by midway through final design (9/91) so that the results may be included in the FSAR.

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position would be to rely on analysis and the current data base. One major consequence of nonexecution may be difficulty in obtaining licensing approval. Secondly, if fluctuations occur after reactor startup, serious consequences will result in terms of costs and schedule delays.

REV

M.C. Hensley 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

concurrence for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/16/87

CONFIRM ABSENCE OF FUEL/REFLECTOR COLUMN VIBRATIONS
DDN M.10.18.20
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Confirmation is required that there are no significant flow-induced vibrations of the fuel/reflector columns in the MHTGR annular prismatic core.

1.1 Summary of Function/Title/Assumptions

F1.1.2.1.2.2.4 "Maintain Fuel Element Structural Integrity," Assumption 12. Flow-induced vibrations do not contribute significantly to fuel element stresses.

1.2 Current Data Base Summary

Multi-column flow tests were performed in 1980 on an eight-block high core design. These tests showed that the columns were susceptible to flow-induced vibration at certain flow rates.

1.3 Data Needed

Data are required to confirm that the fuel/reflector columns have no significant flow induced-vibrations under any reactor operating conditions.

a. Gap coolant velocities, m/s (ft/s)

Minimum	[TBD (TBD)]
Maximum	[TBD (TBD)]

b. Coolant densities, kg/m^3 (lbm/ft^3)

Minimum	[0 (0)]
Maximum	[0.36 (28)]

c. Gap Flow, kg/s (lbm/s)

Minimum	[TBD (TBD)]
Maximum	[TBD (TBD)]

2. DESIGNER'S ALTERNATIVES

2.1 Rely on current design analysis methods and data base.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to validate the design with experimental data. The current data base is not considered adequate for the present 10-block high core design. Flow analyses are not considered significantly accurate to provide the necessary confidence that the core design will not be susceptible to flow-induced vibrations.

4. SCHEDULE REQUIRMENTS

Data are needed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to rely on analysis and the current data base. The consequence of nonexecution could be that it would be necessary to redesign the core restraint system, if excessive vibration were encountered at reactor startup. This would result in schedule delays and cost increases.

REV

M. C. Henderson 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/11/87

RESERVE SHUTDOWN AND BURNABLE POISON MATERIAL PROCESS DEVELOPMENT
DDN M.10.18.21
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The B_4C granules dispersed in the lumped burnable poison (LBP) rods and reserve shutdown control (RSC) pellets will be coated with pyrocarbon (PyC) to protect the B_4C from oxidation during normal service and postulated moisture and air ingress events. The B_2O_3 phase which would form without the protective PyC coating is not refractory and could volatilize at temperatures as low as 300°C under moist conditions. Such a condition could lead to loss of reactivity control. Processes are needed to produce the PyC coated B_4C granules and form them into the boronated graphite compacts without damage to the coating.

1.1 Summary of Function/Title/Assumptions

F1.1.1.2.2.2 "Absorb in Fixed Poisons," Assumption 1: Processes are available to deposit a pyrocarbon coating on the B_4C granules in the lumped burnable poison compacts to protect them from chemical attack.

F3.1.1.2.1.1.2.1.1 "Control with Movable Poisons," Assumption 1: Processes are available to deposit a pyrocarbon coating on the B_4C granules in the reserve shutdown pellets to protect them from chemical attack.

1.2 Current Data Base Summary

The current data base for boronated graphite oxidation consists of results from tests on uncoated B_4C dispersed in graphite. Those tests confirm the preferential oxidation of B_4C to form B_2O_3. The B_2O_3 layer forms around each B_4C granule and then the rate of reaction slows unless the B_2O_3 layer is removed by vaporization. The boiling temperature for the B_2O_3 layer is 1250°C so vaporization near peak normal condition temperature would be rapid even in dry helium. In moist helium the B_2O_3 reacts with water to form boric acid (H_3BO_3) which is volatile at temperatures as low as 300°C.

The design curves currently used in MHTGR design indicate that with a moisture level of 1000 ppm, 50 atm total pressure, and a temperature of 650°C, the rate of boron oxidation would approach 10% per day of exposure. Since the time of high moisture events could be as high as 40 days, the predicted extent of boron oxidation is very high.

In capsule R2-K13 B_4C particles of about 200 μm diameter were coated with PyC and dispersed as a monolayer in graphite wafer compacts. The wafers were placed in the capsule and irradiated to high burnup and fluence at 1000° and 1200°C, well beyond requirements for the MHTGR. The B_4C was coated first with a buffer and then a dense PyC coating to produce a BISO coating. The material performed very well with no coating failure or adverse dimensional change effects.

The basic material oxidation data indicate that the PyC is much less reactive than the B_4C and the graphite matrix. Therefore, encapsulating the B_4C in PyC is expected to provide the desired protection under normal conditions and less than the most severe accidents. However, under the most severe moisture ingress events the PyC coating may not provide adequate protection and the alternative of SiC coating may be needed.

1.3 Data Needed

A process is needed to deposit a coating of PyC on B_4C granules. The granule size must cover a range appropriate for the reserve shutdown and the burnable poison. Granule diameter of 300 to 500 μm is anticipated. The coating thickness must be established so that it is thick enough to provide the desired oxidation protection but not so thick that the required boron loading can not be obtained with current compact-making technology. Prior experience with coated B_4C has been with spherical material, but for economic reasons the desired material form is fragmented as it comes from the crushing of B_4C ingots.

The process development program must establish the allowable particle size, density, shape, coating batch size, and coating conditions producing an acceptable coating. It is anticipated that a buffer coating will not be needed so only the equivalent of the OPyC coating of the reference fuel coating (TRISO) will be deposited and characterized.

The program will produce process and product specifications for PyC coated B_4C for both the reserve shutdown and burnable poison boronated graphite materials. Quality Assurance must be in accordance with requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Coatings for B_4C are needed which will provide the oxidation protection and be compatible with existing processes for manufacture of the following control materials:

1.4.1 Reserve Shutdown Control (RSC) Pellets

- o Boron Loading $\leq$40 wt %
- o Spherical Pellet Diameter 10 mm
- o Pellet Density $\geq$1.5 mg/m^3

1.4.2 Lumped Burnable Poison (LBP) Compact

- o Boron Loading $\leq$3 wt %
- o Diameter 11 mm
- o Length 50 mm
- o Compact Density $\geq$1.7 mg/m^3

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Coat the B_4C granules with SiC.

2.2 Instead of coating the B_4C granules with PyC or SiC, coat the entire RSS pellet or burnable poison compact with PyC to provide the needed oxidation protection.

2.3 Instead of B_4C, use oxidation resistant poisons such as rare earths, gadolinium, or hafnium oxides.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The use of PyC-coated B_4C was selected because it promises a high probability of suitable protection against oxidation while still providing the favorable neutronic properties of boron. The use of a PyC coating eliminates the need for extensive process development since existing coating technology can be used. The thermal stability and irradiation performance of B_4C in graphite has already been established and the use of coated B_4C would not alter physical properties significantly. Therefore, much of the existing data could be used to qualify the material for use in a reactor core.

Alternatives 2.1, 2.2, 2.3 were not chosen as the reference design because they represented higher costs for fabrication and qualification for use in the MHTGR.

4. SCHEDULE REQUIREMENTS

The process must be demonstrated before completion of preliminary design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty of existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to coat the B_4C granules with SiC. The process parameters used for coating fuel with SiC would be modified to coat the B_4C so the coating development effort would not be large. However, the coating time for SiC is about 10X longer than for PyC so the manufacturing cost will be higher for SiC.

O. M. Stansfield 3/17/87

Originator Date

R. F. Turner 3/17/87

Department Manager Date

Ollisaan for
G. C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 3/27/87

CORROSION CHARACTERISTICS OF COATED B_4C
DDN M.10.18.22
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 18

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The pyrocarbon-coated B_4C granules in the lumped burnable poison (LBP) compacts of the fuel elements and in the reserve shutdown control (RSC) pellets may be corroded by coolant impurities, principally H_2O, which would compromise the reactivity control capability. The exposed B_4C may hydrolyze and the resulting boric acid, which is quite volatile, may be lost from the core. Therefore, the corrosion characteristics of the pyrocarbon-coated B_4C granules must be quantified for normal operating conditions and for H_2O ingress events.

The rate of oxidation of B_4C granules by H_2O or O_2 is rapid above 800°C, and the process is mass transfer limited. This condition has prompted the need for protective coatings, such as PyC or SiC, similar to the coatings on fuel particles. The use of PyC will offer protection of the B_4C granules, with time limits for long-term exposure to steam or air at elevated temperatures (up to 1600°C), such as during the combined heatup and moisture or air ingress accident.

1.1 Summary of Function Number/Title/Assumptions

F1.1.1.2.2.2 "Absorb in Fixed Poisons," Assumption 2: Reference correlations for the corrosion of lumped burnable poison compacts by coolant impurities are accurate to within a factor of [2] at 95% confidence.

F2.1.1.2.2.2 "Protect the Capability to Absorb in Fixed Poisons," Assumption 1: Reference correlations for the corrosion of lumped burnable poison compacts by coolant impurities are accurate to within a factor of [2] at 95% confidence.

F3.1.1.2.1.1.2.1.1 "Control with Movable Poisons," Assumption 2: Reference correlations for the corrosion of reserve shutdown pellets by coolant impurities are accurate to within a factor of [2] at 95% confidence.

F3.1.1.2.1.1.2.1.3 "Control with Fixed Poisons," Assumption 1: Reference correlations for the corrosion of lumped burnable poison compacts are accurate to within a factor of [2] at 95% confidence.

1.2 Current Data Base Summary

Experiments in which irradiated BISO coated fuel particles inside graphite crucibles were exposed to 17% steam in He at 1200°C showed the PyC coatings were breached in 1 to 2 hours. Unirradiated particles at the same conditions failed in about 10 h after much of the surrounding graphite was consumed. At temperatures below about 1000°C, rupture times were longer, 20 to 40 hours, and were unaffected by the sacrificial oxidation of surrounding graphite. The consequences of failed coating would be rapid oxidation of the B_4C, forming B_2O_3, which could then be vaporized out of the particle and the core.

1.3 Data Needed

Correlations are needed describing the corrosion of PyC-coated B_4C granules by coolant impurities during normal operating conditions and H_2O ingress events. Data are needed to characterize the reaction kinetics; the reaction rate must be determined as a function of temperature, impurity concentrations, system pressure, and time. In addition, the possible effects of radiolysis and catalysis by impurities and by fission metals on the reaction kinetics must be determined. The temperature below which the oxidation reaction is not mass-transfer limited must be confirmed.

Finally, evidence must be provided to show that in the integral system containing LPB compacts, RSS pellets, H-451 graphite, and fuel compacts, the models used to predict oxidation of the poison material and boron loss from the core have the required accuracy.

Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation (LBP Compacts)

Environment	Helium
Maximum Fast Fluence (E >29 fJ)	5×10^{25} n/m^2
Maximum Gamma Flux	[TBD] MeV/cm^2-s
Temperature Range	[300 - 1200] °C

Coolant Impurity Levels	126 µatm H_2O 315 µatm CO 126 µatm CO_2 Total Oxidants <630 µatm 630 µatm H_2
Helium Coolant Pressure	>[10] atm

Normal Operation (RSC Pellets)

Environment	Helium
Maximum Fast Fluence (E >29 fJ)	[TBD] n/m²
Maximum Gamma Flux	[TBD] MeV/cm²-s
Temperature Range	[300 - TBD] °C
Coolant Impurity Levels	126 µatm H_2O 315 µatm CO 126 µatm CO_2 Total Oxidants <630 µatm 630 µatm H_2
Helium Coolant Pressure	>[10] atm

Moisture Ingress Conditions
(LBP Compacts & RSC Pellets when in Core)

Environment	He/H_2O
Coolant Pressure Range	>[10] to 1 atm
Range of Coolant Impurity Levels	[0.01 - 1.0] atm H_2O
Temperature Range	[300 - 1200] °C

Moisture Ingress Conditions (RSC Pellets when in Hoppers)

Environment	He/H_2O
Coolant Pressure Range	>[10] to 1 atm
Range of Coolant Impurity Levels	[0.01 - 1.0] atm H_2O
Temperature Range	[TBD] °C

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use SiC-coated B_4C granules.

2. Use the current data base and add more margin to account for the uncertainties.

3. Impose tighter technical specifications on primary coolant oxidant levels.

4. Use more corrosion resistant rare earth control materials.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Determine the corrosion characteristics of PyC-coated B_4C granules under normal operating and H_2O ingress conditions relevant to the 350 MW(t) Modular HTGR. The uncertainties in the current data base would necessitate unacceptably large design margins. Imposition of tighter tech spec limits on coolant impurities is expected to adversely impact plant availability. The use of SiC-coated granules in the LBP compacts would lead to a neutronic penalty, and the deposition of a SiC coating on the B_4C granules would be more costly than PyC coating. Development and qualification of rare earth control materials would add significant development costs.

4. SCHEDULE REQUIREMENTS

Preliminary data by 3/89 (6 months prior to PSSAR); and final data by 9/92 (one year prior to FSSAR submittal).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Alternative 1 is the fallback position with the attendant need to develop and qualify SiC-coated B_4C. The consequence would be more expensive process development and product qualification programs along with the attendant neutronic penalty for adding Si to the core. The use of SiC-coated B_4C would be pursued only if the PyC-coated B_4C proved not to be sufficiently corrosion resistant.

DR Hanson 3/24/87
Originator Date

RJ Turner 3/24/87
Department Manager Date

for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/27/87

CORROSION CHARACTERISTICS OF CORE MATRIX MATERIALS
DDN M.10.18.23
PROJECT NUMBER 6300

PLANT: 4 × 350 MW(t) Modular HTGR/System 18

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The carbonaceous matrix materials used as binders in the fuel compact, in the lumped burnable poison (LBP) compact, and the reserve shutdown control (RSC) pellets consists of finely divided graphite flakes bonded together with residual carbon from the carbonized pitch binder. The function of the matrix is to provide a stable, refractory bond between components such as fuel particles or poison materials. The matrix may be corroded by coolant impurities, principally H_2O. Under certain circumstances, these matrix materials may serve as getters and protect the fuel particles and B_4C control materials from corrosive agents. However, if the corrosion of these materials is extensive, there could be deleterious effects. For example, corrosion could potentially lead to loss of structural integrity for the fuel compact with reduced thermal conductivity and associated higher temperatures. In the B_4C containing compact, complete corrosion of the matrix would decrease the height of the B_4C column in the core and make control of the core more difficult. Therefore, the corrosion characteristics of these matrix materials must be quantified for normal operating conditions and for H_2O ingress events.

1.1 Summary of Function Number/Title/Assumptions

F1.1.1.2.2.2 "Absorb in Fixed Poisons," Assumption 2: Reference correlations for the corrosion of lumped burnable poison compacts by coolant impurities are accurate to within a factor of [2] at 95% confidence.

F2.1.1.2.2.2 "Protect the Capability to Absorb in Fixed Poisons," Assumption 1: Reference correlations for the corrosion of lumped burnable poison compacts by coolant impurities are accurate to within a factor of [2] at 95% confidence.

F3.1.1.2.1.1.2.1.1 "Control with Movable Poisons," Assumption 2: Reference correlations for the corrosion of reserve shutdown pellets by coolant impurities are accurate to within a factor of [2] at 95% confidence.

F3.1.1.2.1.1.2.1.3 "Control with Fixed Poisons," Assumption 1: Reference correlations for the corrosion of lumped burnable poison compacts are accurate to within a factor of [2] at 95% confidence.

F1.1.4.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 8: Reference correlations are adequate to describe the corrosion of fuel-compact matrix to within factor of [2] at 95% confidence.

F2.1.4.1.1.2.1.1 "Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption 6: Reference correlations are adequate to describe the corrosion of fuel-compact matrix to within factor of [2] at 95% confidence.

F3.1.1.2.1.1 "Retain Radionuclides in Fuel Particles," Assumption 6: Reference correlations are adequate to describe the corrosion of fuel-compact matrix to within factor of [2] at 95% confidence.

1.2 Current Data Base Summary

The corrosion of matrix materials used in the fuel compact, in the lumped burnable poison (LBP) compact, and the RSC pellets by coolant impurities (H_2O, O_2, and CO_2) may be mass-transfer limited, chemical-reaction limited, or a combination of both; consequently, both processes must be characterized. The transport of coolant impurities in these materials is by pore diffusion (and by convection as well in the case of the fuel compact matrix), and the transport rates may increase with increasing matrix burnoff. This transport process is characterized by an effective diffusion coefficient. The process is the same as in graphite where the effective diffusion coefficients have been obtained for H-327 and H-451 graphite. However, the permeability of these matrix materials has not been well characterized.

The reference correlations for the kinetics of fuel compact matrix corrosion by coolant impurities are based primarily upon laboratory measurements on small unirradiated specimens in helium with high impurity levels at or near atmospheric pressure. The results of one study in 1975 showed there was little difference in oxidation rate between matrix, PyC or graphite. In earlier tests in 1974 pure matrix was more reactive than H-451 by a factor of 20. Since the measurements were all made on unirradiated matrix, the effects of radiolysis and catalysis by impurities or fission metals on the fuel compact corrosion rate were not systematically investigated. The reaction of H_2O with H-451 graphite exhibits enhancement by Langmuir-Hinshelwood type kinetics with significant reaction inhibition by H_2 but not by CO. The kinetics of fuel-compact matrix corrosion is assumed to be similar to H-451 graphite, but the rate is assumed to be 10x higher.

1.3 Data Needed

Correlations describing the corrosion of core matrix materials by coolant impurities during normal operation and H_2O ingress events are needed. Data are needed to characterize both the transport of coolant impurities and corrosion products in these materials and the intrinsic kinetics for the reaction of water and oxygen with them. To characterize the transport of coolant impurities, the porosity, tortuosity, and permeability must be determined. To characterize the reaction kinetics, the reaction rate must be determined as a function of temperature, impurity concentrations, system pressure, and time. In addition, the effects of radiolysis and catalysis by matrix impurities and by fission metals on the reaction kinetics must be determined. Finally, the effects of partial matrix burnoff on both the mass transfer processes and the intrinsic reaction kinetics must be quantified.

Quality assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The service conditions of interest are given below.

Normal Operation (Fuel Compacts)

Environment	Helium
Maximum Fast Fluence (E >29 fJ)	5×10^{25} n/m^2
Maximum Gamma Flux	[TBD] MeV/cm^2-s
Temperature Range	[300 - 1250] °C
Coolant Impurity Levels	125 μatm H_2O 315 μatm CO 126 μatm CO_2 Total Oxidants 630 μatm 630 μatm H_2
Helium Coolant Pressure	>[10] atm

Normal Operation (LBP Compacts)

Environment	Helium
Maximum Fast Fluence (E >29 fJ)	5×10^{25} n/m^2
Maximum Gamma Flux	[TBD] MeV/cm^2-s
Temperature Range	[300 - 1200] °C

Range of Coolant Impurity Levels	126 µatm H_2O 315 µatm CO 126 µatm CO_2 Total Oxidants 630 µatm 630 µatm H_2
Helium Coolant Pressure	>[10] atm

Normal Operation (RSC Pellets)

Environment	Helium
Maximum Fast Fluence (E >29 fJ)	[TBD] n/m^2
Maximum Gamma Flux	[TBD] MeV/cm^2-s
Temperature Range	[300 - TBD] °C
Coolant Impurity Levels	[126] µatm H_2O [315] µatm CO [126] µatm CO_2 Total Oxidants [630] µatm [630] µatm H_2
Helium Coolant Pressure	>[10] atm

Moisture Ingress Conditions
(Fuel Compacts, LBP Compacts & RSC Pellets when in Core)

Environment	He/H_2O
Coolant Pressure Range	>[10] to 1 atm
Range of Coolant Impurity Levels	[0.01 - 1.0] atm H_2O
Temperature Range	[300 - 1200] °C

Moisture Ingress Conditions (RSC Pellets when in Hoppers)

Environment	Helium
Coolant Pressure Range	>[10] to 1 atm
Range of Coolant Impurity Levels	[0.01 - 1.0] atm H_2O
Temperature Range	[TBD] °C

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

1. Use the current data base and add more margin to account for the uncertainties.

2. Impose tighter tech specs on primary coolant oxidant levels.

3. Use higher purity, more corrosion resistant matrix materials.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Determine the corrosion characteristics of core matrix materials under normal operating and H_2O ingress conditions relevant to the 350 MW(t) Modular HTGR. The uncertainties in the current data base would necessitate unacceptably large design margins. Imposition of tighter tech spec limits on coolant impurities is expected to adversely impact plant availability. Development and qualification of a higher purity matrix materials would add significant development costs.

4. SCHEDULE REQUIREMENTS

Preliminary data by 3/89 (6 months prior to PSSAR submittal); and final data by 9/92 (one year prior to FSSAR submittal).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

A combination of Alternatives 1 and 2 with the necessity of added margins in the design to compensate for uncertainties in the extent of matrix corrosion. The consequence would be unnecessarily restrictive tech specs on primary coolant impurity levels which could have a very adverse impact on plant availability.

D L Hanson 3/24/87
Originator Date

R F Turner 3/24/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/16/87

VALIDATE THE PRESSURE DROP AND FLOW MIXING IN THE
LOWER REFLECTOR/CORE SUPPORT BLOCKS
DDN M.10.18.24
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Confirmation is required that the following limits are met: core pressure drop, maldistribution of coolant channel flows in the columns, the temperature of the coolant entering the hot duct, and the temperature of hot and cold streaks entering the steam generator.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.1.1.1.3.2 "Channel Primary Coolant Through Core," Assumption 2: Pressure loss coefficients through lower reflector and core support blocks are valid.

F1.1.2.1.2 "Transfer Heat from Heat Transfer Surface to Primary Coolant," Assumption 4: Flow geometries in the metallic elements/top reflector, lower reflector/core support, and core upper and lower plenums do not significantly affect the core coolant channel flow distribution.

F1.1.2.2.1.1.1.3.2 Channel Primary Coolant Through Core," Assumption 3: Coolant temperature attenuation coefficients in the lower reflector and core support are valid.

1.2 Current Data Base Summary

No data for the pressure drop, flow distribution, or mixing through the lower reflector and core support blocks exist. Pressure loss coefficients are currently estimated from data available in the general literature.

1.3 Data Needed

Pressure drop, coolant channel flow distribution, and hot/cold streak attenuation data as a function of Reynolds number and location of hot/cold streaks. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Reynolds number range

Maximum	[220,000]
Minimum	[0]

b. Hot/cold streak location [TBD]

2. DESIGNER'S ALTERNATIVES

2.1 Use the loss coefficient estimates based on data in the general literature. Estimate mixing and flow distribution.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to determine the pressure loss coefficients, flow distribution, and hot/cold streak attenuation by testing. Relying on estimates based on available data and engineering judgment would provide little confidence in the results. Also, testing may identify modifications to reduce pressure drop and/or improve flow distribution or hot/cold streak attenuation.

4. SCHEDULE REQUIREMENTS

By the end of preliminary design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Use estimates based on data in the general literature and engineering judgment. Consequences are a less optimized design; allow for higher core pressure drop in the circulator design, higher hot/cold streaks exiting the core, and greater flow maldistribution in the fuel columns.

REV

M. C. Henderson 3/17/87
Originator Date

R J Turner 3/17/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 3/16/87

VALIDATE THE PRESSURE DROP AND FLOW DISTRIBUTION
THROUGH THE METALLIC PLENUM ELEMENT AND TOP REFLECTOR
DDN M.10.18.25
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) Modular HTGR/System 10

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Confirmation is required that the following limits are met: core pressure drop and maldistribution of the coolant channel flows in the columns.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.1.1.1.3.2 "Channel Primary Coolant Through Core," Assumption 1: Pressure loss coefficients through metallic elements at the top reflector are valid.

F1.1.2.1.2 "Transfer Heat from Heat Transfer Surface to Primary Coolant," Assumption 4: Flow geometries in the metallic elements/top reflector, lower reflector/core support, and core upper and lower plenums do not significantly affect core coolant channels flow distribution.

1.2 Current Data Base Summary

No data exist for the flow geometry through the metallic elements at the top of the core. Estimates for loss coefficients have been made based on data available in the general literature.

1.3 Data Needed

Column average loss coefficient and local channel flow distribution as a function of channel Reynolds number. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

a. Reynolds number range

Maximum	[TBD]
Minimum	[TBD]

2. DESIGNER'S ALTERNATIVES

2.1 Use the estimates based on data in the general literature.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to determine the pressure loss coefficients and flow distribution by testing. Estimates based on available data for simple flow geometries are highly uncertain. Also, the test may identify modifications to reduce pressure drop and/or improve flow distribution.

4. SCHEDULE REQUIREMENTS

By end of preliminary design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

Use estimates based on data in the general literature. Consequences are a less optimized design; allow for higher core pressure drop in the circulator design, higher hot/cold streaks exiting the core, and greater flow maldistribution in the fuel columns.

REV

M.C. Hendee 3/17/87
Originator Date

R F Turner 3/17/87
Department Manager Date

[illegible] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

NIL-DUCTILITY TRANSITION TEMPERATURE SHIFT FOR REACTOR VESSEL MATERIAL IRRADIATED AT LOW TEMPERATURES
DDN M.11.06.01
PROJECT NUMBER 6300

Plant: 4 X 350 MW(t) MHTGR/System 11

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insufficient data base for determining the neutron-induced nil-ductility transition temperature (NDTT) shift, exists for the reactor vessel material (SA533 Grade B, Class 1), weldment, and heat-affected zone.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.2.1.1, "Maintain Primary Coolant Boundary Integrity," Assumption 2: The LWR data base for NDTT shift is applicable to the MHTGR reactor vessel fluence spectrum. The LWR data base for NDTT shift can be extrapolated to the low operating temperatures of the MHTGR reactor vessel. Assumption 3: A sufficient data base to calculate reactor vessel NDTT shift will be available.

F1.1.2.2.2.1.1.1, "Maintain Material Strength," Assumption 3: A sufficient data base to calculate reactor vessel NDTT shift will be available.

F1.2.2.2.2.1.1, "Maintain Primary Coolant Boundary Integrity," Assumption 3: A sufficient data base to calculate the reactor vessel NDTT shift will be available.

F1.4.2.2.2.1.1, "Maintain Primary Coolant Boundary Integrity," Assumption 3: A sufficient data base to calculate the reactor vessel NDTT shift will be available.

F2.1.2.2.2.1.1, "Protect the Capability to Maintain Primary Coolant Boundary Integrity," Assumption 2: A sufficient data base to calculate the reactor vessel NDTT shift will be available.

F2.2.2.2.2.1.1, "Protect the Capability to Maintain Primary Coolant Boundary Integrity," Assumption 2: A sufficient data base to calculate the reactor vessel NDTT shift will be available.

F2.4.2.2.2.1.1, "Protect the Capability to Maintain Primary Coolant Boundary Integrity," Assumption 2: A sufficient data base to calculate the reactor vessel NDTT shift will be available.

1.2 Current Data Base Summary

The pressure vessel integrity of operating commercial LWR power plants is based on a large data base of material property evaluations, fracture mechanics evaluations, and engineering experience in vessel design and fabrication. The reference temperature (RT_{NDT}) concept was introduced into the ASME Boiler and Pressure Vessel Code (Section III, NB-2300) in 1972. It provided a definitive and conservative method for specifying materials and for establishing reactor vessel operating limits. Criteria for reactor vessel material toughness (including RT_{NDT}) and design of surveillance (monitoring) programs for vessel beltline materials were established in the Code of Federal Regulations in 1973 (10CFR50 Appendices G & H). In 1975, Regulatory Guide 1.99 was issued, providing the official NRC position on prediction of radiation effects for reactor vessel materials. Revision 02 of this Regulatory Guide is currently under review by NRC. Current research concentrates on the refinement of radiation damage mechanistic models for use in predicting radiation damage.

The bulk of the data is from and applicable to typical LWR conditions: irradiation temperatures in the range 530 - 580°F; total neutron fluxes in the range 10^{10} - 10^{13} n/cm^2.s; and the spectrum for fast, epithermal, and thermal neutrons of 10%, 25%, and 65%, respectively. Analyses of the shift data have not identified an effect of varying irradiation temperature within the narrow band. However, limited data are available outside the range to indicate that lowering the irradiation temperature results in a significant increase in damage. The data are correlated with fast neutron fluence as evidenced by the procedures in Reg. Guide 1.99, Rev. 02. To specifically account for the effect of neutron spectrum, the equivalent fast fluence approach can be employed to allow the use of the procedures in Regulatory Guide 1.99.

1.3 Data Needed

Data are needed to characterize the neutron-induced NDTT shift in the reactor vessel material at irradiation temperatures (300-420°F) and neutron flux, fluence, and spectrum levels expected for the MHTGR. The locations of concern for which data are needed include the reactor vessel beltline and closure head. The data need to be sufficient to meet a 95% confidence that these properties meet or exceed design values. Quality Assurance must be in accordance with requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

The results of this program are expected to provide the designer a means of calculating a conservative value for NDTT shift through a correlation with equivalent fast fluence.

The specified service conditions for the reactor vessel are as follows.

Normal Operation

RV Beltline

- 398°F (203°C) to 418°F (214°C) metal temperature
- 925 psia (6.38 MPa)
- 1.8×10^{17} n/cm^2 total fluence

  2.6×10^{16} n/cm^2 equivalent fast fluence
- 280 000 h (40 yr at 80% availability)

RV Closure Head

- 300°F (149°C) to 390°F (199°C) metal temperature
- 925 psia (6.38 MPa)
- 5.6×10^{16} n/cm^2 equivalent fast fluence
- 280 000 h (40 yr at 80% availability)

2.0 DESIGNER'S ALTERNATIVE

There are no alternatives. The procedures in Regulatory Guide 1.99 may be used for the prediction of neutron irradiation damage to the reactor vessel when credible surveillance data from the reactor in question are not available. The procedure of the Regulatory Guide do apply to the selected MHTGR reactor vessel material, but for temperatures in the range 525 to 590°F and fast fluence in the range 10^{17} to 10^{20} n/cm^2. Corrections for operations outside this range should be justified by reference to actual data.

3. SELECTED APPROACH AND EXPLANATION

The proposed approach is to obtain a sufficiently large data base for the expected MHTGR operating temperature range and bounding damage fluence. The resulting data will be correlated with equivalent fast fluence. This will provide the designer with a conservative procedure to estimate vessel irradiation damage.

4. SCHEDULE REQUIREMENTS

Interim results on all data are needed, as they became available, by the end of the Conceptual Design phase (9/87). Final results on all data are needed by the end of the Preliminary Design phase (9/89).

5. PRIORITY

Urgency: 1

Cost Benefit: H
Uncertainty in Existing Data: H
Importance of New Data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NON-EXECUTION

A much thicker reactor vessel would keep the operating stresses very low so that an adequate margin to brittle failure is maintained (according to the procedure of the ASME Code Section III NB-2300). A much more extensive material surveillance program would be needed to monitor the vessel material. Furthermore, to reduce the uncertainty with regard to pressurized thermal shock, additional plant protection control modes may be necessary. The effects of these measures would be increased costs.

Consequences of program non-execution are:

- The NRC would not accept the design basis and would not allow operation.
- This is a safety area in which a brittle fracture could lead to a sudden break of a primary coolant boundary.

Gerald de Lauw 2/27/87
Originator Date

C.H. Amper 2/27/87
Department Manager Date

A. McEwinty 2/27/87
Manager, Project Operations Date

DETERMINE PROPERTIES OF SA533B
(Mn-1/2Mo - 1/2 Ni) BASE METAL AND WELDMENT
AT ELEVATED TEMPERATURES
DDN M.11.06.02
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) MHTGR/System 11

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insufficient property data base exists for the design of the reactor vessel fabricated from SA533 Grade B, Class 1 material and associated weldment which are exposed to elevated temperatures (> 700°F).

1.1 Summary of Function/Title/Assumptions

F2.1.2.2.2.1.1, "Protect the Capability to Maintain Primary Coolant Boundary Integrity," Assumption 1: A sufficient data base will exist to qualify SA533B for Section III of the ASME Boiler and Pressure Vessel Code at elevated temperatures.

F2.2.2.2.2.1.1, "Protect the Capability to Maintain Primary Coolant Boundary Integrity," Assumption 1: A sufficient data base will exist to qualify SA533B for Section III of the ASME Boiler and Pressure Vessel Code at elevated temperatures.

F2.4.2.2.2.1.1, "Protect the Capability to Maintain Primary Coolant Boundary Integrity," Assumption 1: A sufficient data base will exist to qualify SA533B for Section III of the ASME Boiler and Pressure Vessel Code at elevated temperatures.

1.2 Current Data Base Summary

Section III of the ASME Code provides ultimate tensile strength and yield strength data for SA533B to temperature of 1000°F. Section III provides allowable stress intensities for Service Level A, B, C, and D loadings up to a temperature of 700°F derived from these data. Since time-dependent effects (creep) must be considered as temperature increases, allowable design stress intensities are not provided for the elevated temperature regime (> 700°F). Therefore, the derivation and use of allowables for Service Level C and D loadings is not permitted in this elevated temperature regime.

Limited creep strain data are available for this material for temperatures up to 1200°F and for exposures of approximately 48 hours. A constitutive equation has been fitted to the data. Elastic-plastic properties are available for temperatures up to 1200°F and for strains up to 5%. Analytical relationships were developed for the stress vs plastic strain.

1.3 Data Needed

Data are needed to support a Section III Code inquiry to establish allowable stress intensities for SA533 Grade B, Class 1 and its weldment for Service Level C and D loadings at elevated temperatures. To satisfy this objective creep strain rate data are needed. Demonstration of negligible creep for the anticipated service conditions would permit the extension of the Section III rules for special event (Service Level C and D) allowables to the 700°F to 1000°F temperature range. If significant creep strains are encountered, then isochronous stress-strain curves would also be required for the anticipated service conditions and duration. Special event allowable stress intensities would then be established. These data need to be sufficient to meet a 95% confidence that these properties meet or exceed design values. Quality Assurance must be in accordance with the requirements for Quality Assurance Level I.

1.4 Data Parameters/Service Conditions

Data are needed to establish allowable stress intensities for SA533 Grade B, Class 1 and its weldment for temperatures from 700°F to 900°F and load durations up to 400 hours.

The specified service conditions for the reactor vessel are as follows:

Normal Operation

- Metal temperature < 700°F (371°C)
- 280 000 hours (40 yr. at 80% availability)
- Service Environment:
  Helium with 0.5 ppmv H_2O, 3 ppmv $CO + CO_2$, 3 ppmv H_2, 0.1 ppmv CH_4, and 2 ppmv N_2 on the internal surface.

  Air in Reactor Building on the external surface.

Off-Normal Operation

- Service Level B
  Metal temperature < 700°F (371°C)
- Service Level C
  Pressurized Conduction Cooldown Event
  Maximum metal temperature 764°F (407°C)
  Time duration above 700°F (371°C) ca. 250 hours
  Maximum hoop membrane stress 25 ksi (170 MPa)

- Service Level D

  Pressurized Conduction Cooldown Event
  Maximum metal temperature 771°F (411°C)
  Time duration above 700°F (371°C) ca. 250 hours
  Maximum hoop membrane stress 25 ksi (170 MPa)

  Depressurized Conduction Cooldown Event
  Maximum metal temperature 880°F (471°C)
  Time duration above 700°F (371°C) ca. 400 hours.
  Shell axial stress < 1 ksi (7 MPa)

2. DESIGNER'S ALTERNATIVES

The alternatives are as follows:

2.1 A sufficient data base should be established to qualify SA533 Grade B, Class 1 for Code Case N47.

2.2 A Code Case N47 approved material could be selected for the Vessel System. The current choices are 2-1/4 Cr - 1 Mo, 304SS, 316SS, and Alloy 800H.

2.3 Active, high-reliability systems can be incorporated which will insure maintaining the vessel temperature below 700°F.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to apply the highly reliable and highly developed LWR technology to the reactor vessel. This includes the application of SA533 Grade B, Class 1 with its extensive irradiation data base which needs only an incremental expansion. The ASME Code design procedures and associated supporting data and allowable design stress intensities are applicable to virtually the entire duty cyle with the exception of several Service Level C and D events. In these events the vessel material exceeds 700°F, the upper limit for which allowable design stress intensities are available in Section III of the ASME Code. The proposed approach would permit the establishment of allowable stress intensities using existing Section III data and rules. At worst, if creep were significant, methods and procedures would be developed to establish allowables from the data.

It is judged that developing a special Code Case for SA533 Grade B, Class 1, applicable to at least 900°F is the most cost-effective solution.

Alternative 2.1 to make SA533 Grade B, Class 1 Code Case N47 approved material would involve the establishment of a large data base including: time-dependent allowable stress intensities, creep strain rates, isochronous stress-strain curves, fatigue strain range curves, and creep-fatigue damage envelope. The range of data would go far beyond the limited needs to support the reactor vessel design. This approach would require a DDN which is expected to be much more expensive.

Alternative 2.2 would involve selecting a material which does not have an irradiation data base with regard to ductility. A very extensive data base comprising chemistry effects as well as temperature and fluence, would be required. This approach also requires a DDN which is expected to be more costly than the selected approach.

Alternative 2.3 would result in increased capital cost for the nuclear island, and there would be a loss in the simplicity of design and operation.

4. SCHEDULE REQUIREMENTS

Interim results on all data are needed, as they become available, by the end of the Conceptual Design phase (9/87). Final results on all data are needed by the end of the Preliminary Design phase (9/89).

5. PRIORITY

Urgency: 1
Cost Benefits: H
Uncertainty in Existing Data: H
Importance of New Data: H

6. FALLBACK POSITION AND CONSEQUENCE OF NON-EXECUTION

The fallback position is to provide additional, active systems to insure that metal temperatures do not exceed 700°F during all events (i.e., Alternative 2.3).

Gerald A. Lauro 2/27/87
Originator Date

C.H. Casper 2/27/87
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A. McEachern 2/27/87
Manager, Project Operations Date

DATE: 2/27/87

REACTOR EQUIPMENT SERVICE FACILITY TOOLS DESIGN VERIFICATION
DDN M.20.16.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 20

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The multi-purpose machines, and equipment which perform the scheduled remote maintenance and service operations must accomplish their functions within a facility of specified size and layout.

1.1 Summary of Function/Title/Assumptions

F1.1.9 "Maintain Plant Performance Capability," Assumption 1: Maintaining performance capability is a continuing after-design activity that is impacted by each layout, system component, and feature incorporated in the plant design.

1.2 Current Data Base Summary

The conceptual design is based on experience gained from the Fort St. Vrain Hot Service Facility. However, some of the service equipment, machines and tools such as the neutron control assemblies and FHM are unique to the proposed concept because of the taller core and other design changes. Therefore, no data exist for these unique features.

1.3 Data Needed

Data are needed to confirm the adequacy of the remote maintenance features built into the fuel handling equipment, the neutron control assemblies and other reactor equipment requiring remote maintenance. Data on tolerances and compatibility must be gathered. Also, data on the capability of the proposed tool designs to perform the required work in a timely manner must be collected. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal atmosphere	Air
Pressure	14.7 psia
Temperature	120°F
Radiation	[TBD]

2. DESIGNER'S ALTERNATIVES

2.1 Demonstrate the design adequacy of the tools within the finished facility at the site.

2.2 Do the necessary drafting layouts to depict all handling operations and clearances required within facility.

3. SELECTED APPROACH AND EXPLANATION

It is necessary to assess the feasibility of the unique operations early in the design phase so that necessary changes can be made. Changes may be made to the facility or the service equipment where necessary and achievable, and it may be necessary to alter designs of the equipment to be serviced.

The reactor equipment service tools will be tested in a mock-up of the facility. This will provide early results on the capabilities of the equipment to perform the required service operations, provide early warning of problem areas, and will provide information to finalize the designs of components which must be serviced within the facility. Alternative 2.2, a complete reliance on drafting layout, was not chosen due to the weakness of this approach regarding visibility, equipment deflections, etc. Alternative 2.1 was rejected due to its potential for requiring changes to the components to be serviced after the plant is operating.

4. SCHEDULE REQUIREMENTS

Testing must be completed by 12 months prior to finish of the plant final design phase (9/92).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The preferred fallback position is Alternative 2.1. Should the selected approach not be conducted, there is increased risk of not being able to perform the required maintenance until modifications are made to the already existing tools, facilities or components. This might lead to unscheduled downtime.

Edwin C. Harvey 3/25/87
Originator Date

R J Turner 3/25/87
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

[60-19]

DATE: 2/27/87

RESERVE SHUTDOWN VACUUM TOOL DESIGN VERIFICATION
DDN M.20.16.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 20

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA ON VALIDATION TESTING

The Reserve Shutdown Vacuum Tool (RSVT) must successfully remove the reserve shutdown system pellets from a ten element high active core, which is four elements higher than Fort St. Vrain.

1.1 Summary of Functions/Title/Assumptions

F3.1.1.2.1.1.2.1.1 "Control With Movable Poisons," Assumption 4: A tool is available to remove the boron pellets used in the reserve shutdown system.

1.2 Current Data Base Summary

The RSVT conceptual design is based on existing equipment at Fort St. Vrain. The component has been revised to increase the storage capacity, provide dust filters and accommodate the increased core depth. No data are available on these new features.

1.3 Data Needed

Verification of satisfactory operation and/or identification of areas requiring redesign. Data on blower horsepower, most effective insertion rate for maximum blower life, the need for and the effects of filters, and performance limits in all operating modes, including maximum misalignment of channel with the material insertion holes are needed. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Temperature	240°F
Pressure	14.7 psia
Internal atmopshere	Air or helium
Radiation	[TBD]

2. DESIGNER'S ALTERNATIVES

2.1 Test the equipment at the reactor site.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A test will provide early performance data and opportunity to correct deficiencies, so that the tool will function in all operating conditions. A full scale helium test rig with a variable speed hoist and a full depth core shutdown channel are required. Alternative 2.1 was not chosen because it might interfere with more critical activities at the site late in the schedule.

4. SCHEDULE REQUIRMENTS

Test to be completed 2 years after start of the plant final design phase (9/91).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is Alternative 2.1. Nonexecution of the proposed test would introduce the risk of the tool not being able to remove the shutdown material from the full depth of the reactor core causing a delay in plant startup.

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Originator Date

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DETERMINE CORE EXIT PLENUM AND HOT DUCT FLOW FIELD
DDN M.21.00.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The velocity and temperature distributions downstream of the reactor core during main loop and shutdown cooling system (SCS) operation is important in establishing component performance, e.g., lower hot duct, steam generator, and shutdown cooling heat exchanger (SCHX) performance. The main concern is hot/cold streaks and coolant velocity maldistribution during steady-state operations.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2, "Transport Energy in Primary Coolant," Assumption 1. Methods will be developed for the timely prediction of hot/cold streaks attenuation at the core exit and from the core exit to the steam generator inlet.

Assumption 2. Methods will be developed for the timely prediction of local primary coolant velocity maldistributions.

Assumption 3. Test data will be available for the timely prediction of hot streak mixing, hot helium flow maldistribution in the core lower plenum, duct, and plenum leading to the steam generator bundle.

F2.1.2.2.4.3, "Remove Decay Heat," Assumption 5. A method is available for the prediction of hot/cold streaks attenuation at the core exit and from the core exit to the heat exchanger inlet.

Assumption 6. A method is available for the prediction of local primary coolant velocity maldistributions.

Assumption 7. Test data are available for the prediction of hot streak mixing, hot helium maldistribution in the core lower plenum, and plenum leading to the SCHX bundle.

1.2 Current Data Base Summary

The current understanding of flow behavior and hot/cold streaks in a core outlet plenum has been obtained from air flow tests carried out on a 1/6-scale model of the 2240 MW(t) HTGR-SC/C lower plenum and hot duct configurations (HTGR-85-108). These data are considered largely specific to the 2240 MW(t).

Analytical methods, like the turbulent fluid code COMMIX, are available. This type of code can provide velocity, pressure, and temperature profiles downstream of the core in three dimensions. Because of the complexity of the flow field downstream of the core, some of the turbulent diffusivity models used by these codes have to be validated through tests specific to a given design. Another computer code, PLENUM, is available. This code assesses the attenuation of hot/cold streaks in the core lower plenum based on a set of mixing coefficients specific to the 2240 MW(t) HTGR-SC/C design. A set of new mixing coefficients specific to the 350 MW(t) MHTGR is needed to update this code. Some of the 2240 MW(t) model test data can be used to validate turbulent fluid codes like COMMIX, but the mixing coefficients used for the calculation of hot/cold streak temperatures must be developed specific to the 350 MW(t) configuration.

1.3 Data Needed

The velocity, temperature, pressure, and flow distributions for the primary coolant are needed for main loop operations at the core outlets, the hot duct inlet, the hot duct outlet, and upstream of the steam generator FSH bundle at locations as indicated in DDN M.21.02.08. The velocity, temperature, pressure, and flow distributions for the primary coolant are also needed for shutdown coolant system (SCS) operations at the inlet of the SCS duct and at the shutdow heat exchanger (SCHX) bundle.

Temperature measurements should be done in such a way as to generate a series of mixing coefficients from the core outlets to the steam generator and SCHX bundles.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

The data are required at the following service conditions:

Configuration	o Core outlet flow distribution blocks, core lower plenum, hot duct, and steam generator and SCHX inlet plenums up to the bundles.
Operating conditions	o 25% to 100% power main loop operation. o SCS pressurized and depressurized operation.
Coolant	o Helium.

Pressure	o 925 psi (circulator outlet) main loop operation. o 925 psi to atmospheric pressure SCS operation.
Temperature	o 1268°F average core outlet temperature main loop operation. o 1147°F to 192°F SCS operation.
Helium flow	o 1.25×10^6 lb/h main loop operation. o 92,000 lb/h SCS operation.

2. DESIGNER'S ALTERNATIVES

The following alternative is considered:

2.1 Predict the flow field with present methods incorporating results from previous model tests on other reactor configurations. Design affected components conservatively to increase the margins to plant performance requirements, e.g., use of high temperature materials.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Obtain the data needs from air flow distribution tests on a one-half or one-third scale model of the 350 MW(t) core exit plenum, hot duct and inlet configurations for steam generator and SCHX. The core, steam generator, and SCHX will be modeled by appropriate flow resistances. The selected approach provides realistic flow parameters compared to the alternate approach and has a potential to remove excessive conservatism in the design by reducing uncertainties in the analysis. The test rig described above will be used also to provide the data needs described in DDN M.21.02.08.

4. SCHEDULE REQUIRMENTS

These data are required during the preliminary design phase to remove uncertainties in the design (1988).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

If the selected task is not performed, the fallback position is alternative 2.1, which relies on analysis based on available data from model tests on other reactor configurations and conservative design measures to assure adequate design margins. The consequences are potential for increased cost of hot duct, steam generator, SCHX and circulator components, and/or reduced plant performance.

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Originator Date

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Department Manager Date

Dollison for

G. E. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

MAIN CIRCULATOR MAGNETIC AND CATCHER BEARINGS DESIGN VERIFICATION
DDN M.21.01.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The main circulator magnetic bearings comprise variable strength magnetic fields that suspend the high speed/large mass rotor in position. The rotor and support housing shall have no resonant frequencies throughout the full speed range. In the event of a failure in the magnetic suspension system, catcher bearings are used to support the rotor and are required to withstand at least [20] drops without the need for replacement. Because the main circulator rotor is larger, more massive, and operates at a higher speed, its test results will be applicable to the proposed magnetic and catcher bearing design for the shutdown circulator.

1.1 Summary of Function Number/Title/Assumptions

F.1.1.2.2.1.2.4 "Support Shaft," Assumption 1: Magnetic bearing (including catcher) dynamic properties will be verified.

F.2.1.2.2.1.2.4 "Protect the Capability to Support Shaft," Assumption 2: Reliability of catcher bearings will be verified.

F2.1.2.2.4.3.2.1.2.4 "Support Shaft," Assumption 1: Magnetic bearing (including catcher) dynamic properties will be verified.

1.2 Current Data Base Summary

The design of magnetic bearings in the size and load range of interest is essentially state of the art. S2M (Societe de Mecanique Magnetique), the world's leading manufacturer of magnetic bearings, has some proprietary data under various nonrepresentative conditions. Data on characteristics and performances of active magnetic bearings operating in conditions representative of the main circulator environment have not been established. There are several large (5,000 to 10,000 hp) commercial gas compressors on the market which employ magnetic bearings. MBI has an ongoing catcher bearing test with a 1000 lb rotor at up to 12,000 rpm (MHTGR main circulator has a 6500 lb rotor rotating at up to 6820 rpm). There is a lack of data on the reliability of backup "catcher" bearings to repeatedly support the turning rotor for a limited time when the active magnetic field supporting the rotor is lost. BBC/HRB have an ongoing test of a proprietary catcher bearing design for the HTR-500 concept in FRG.

1.3 Data Needed

Data are required to establish for the reference design (1) static and dynamic axial thrust load capacities, stiffness and damping coefficients of radial bearings for the entire operating speed range, (2) sensitivity of the associated electronic control system to outside disturbances, (3) rotor dynamic response to externally induced unbalance loads occurring in the impeller plane of the rotation, and (4) useful life of catcher bearings versus number of drops from full rpm.

Quality assurance must satisfy QAL II requirements.

1.4 Data Parameters/Service Conditions

Data are required to validate the adequacy of the active magnetic and catcher bearings design for the following main circulator service conditions:

Fluid	Helium: [0.5] ppmv water; [3.0] ppmv hydrogen; [2.0] ppmv nitrogen
Temperature	[100°F - 300°F]
Speed Range	0 to 6820 rpm (constant and transient speed conditions)
Load Range	Axial: 2000 lb downward at nominal (6200 rpm) speed to 6500 lb at 0 rpm Radial: 200 in.-oz unbalance
No. of Drops on Catcher Bearing	[20] minimum

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Obtain FRG proprietary data assuming their availability in a timely manner and establish applicability to GA design.

2.2 Rely on analysis and extrapolation of applicable magnetic bearings test data where available. Accept large uncertainty in design and test prototype machine.

2.3 Perform scale model tests and extrapolate results.

2.4 Use 2240 MW(t) type water-lubricated bearing.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to obtain experimental data on a full scale dummy rotor duplicating the mass/dynamic properties of the actual electric motor rotor and circulator impeller. The dummy rotor will include active magnetic bearings and the mechanical catcher bearings. Data from specifically designed tests conducted in the USA is preferred over foreign proprietary data to facilitate interface, verification, modification and application. The selected approach is more cost effective and has less schedule risk than building and testing a prototype machine if design modifications are necessary. If a scale model is used, complexity in scaling mass/dynamic properties would introduce a high degree of uncertainty in test data.

4. SCHEDULE REQUIREMENTS

Completion of validation test required 1-1/2 years after start of final design (3/91).

5. PRIORITY

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

If magnetic and catcher bearings tests are not performed prior to prototype circulator manufacture and testing, then any major design changes required after prototype tests would cause cost and schedule impact. The fallback position is to use the 2240 MW(t) type water lubricated bearing system which has the potential for water ingress into the primary coolant. The 2240 MW(t) bearing system tested at GA has demonstrated substantial improvement over the FSV system.

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Originator Date

RJ Turner 3/12/87
Department Manager Date

Dalisson for
G.C. Bramblett 3.25.87
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DATE: 2/27/87

MAIN CIRCULATOR PROTOTYPE DESIGN VERIFICATION
DDN M.21.01.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The main circulator assembly including compressor impeller, bearing systems, loop shutoff valve, drive motor, instrumentation and controls must be capable of performing its function during all normal operating conditions and transients with a failure probability $\leq 7 \times 10^{-5}$/h. Data is required to verify the capability of the entire main circulator subsystem to provide adequate primary coolant circulation for various plant operating requirements.

1.1 Summary of Function Number/Title/Asumptions

F1.1.2.2.1.2.2 "Power Circulator," Assumption 2: Submerged rotating diode performance in helium will be verified.

F1.1.2.2.1.2.2 "Power Circulator," Assumption 3: Compressor/shutoff valve performance and interaction will be verified.

F2.1.2.2.1.2 "Protect the Capability to Pump Primary Coolant," Assumption 1: Reliability of the entire main circulator will be verified.

1.2 Current Data Base Summary

Data on helium circulators were primarily derived from component testing performed for Fort St. Vrain and the proposed Delmarva plant. The data base has applicability limited to the design of axial compressors and shutoff valves. Data on active magnetic and catcher bearings should be available prior to design verification of the entire main circulator system. There is no data on the performance characteristics of the current main circulator design and its interactions with the associated external systems and controls.

1.3 Data Needed

Data on the functional capability of the entire main circulator system including motor/control/circulator compatibility are required to verify the reference design. Data needed on the main circulator prototype include: (1) aerodynamic performance of the inlet, loop shutoff valve, compressor impeller, and diffuser; (2) motor thrust bearing performance, (3) overspeed capability; (4) structural integrity of rotating parts and supports; (5) noise levels and frequencies; (6) vibration characteristics and critical speeds; (7) shutdown and hot restart capability including hot soak, and (8) extended duration operation.

Quality assurance must satisfy QAL II requirements.

1.4 Data Parameters/Service Conditions

Data are required to verify the performance and reliability of the main circulator prototype under the following full spectrum range of helium conditions, including part load, startup, hot soak simulation:

	Pressurized	Depressurized (1 Day After Shutdown)
Helium Flow Rate (lb/s)	347	6.94
Outlet Pressure (psia)	925	14.03
Inlet Temperature (°F)	491	190.9
Outlet Temperature (°F)	497	198.7
Pressure Rise (psi)	13.2	0.33
Duration of Testing (h)	[200]	
Hot Soak Temperature (°F)	[600]	
Time at Hot Soak (h)	[6]	

2. DESIGNER'S ALTERNATIVES

The following alternatives to obtaining prototype data are available:

2.1 Rely on subassembly tests.

2.2 Demonstrate performance after installation in vessel during hot flow tests.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to test the complete prototype configuration, including circulator, ducting, loop shutoff valve, service system, instrumentation, motor and control over the entire range of anticipated reactor conditions. Motor/circulator/control interfaces require verification testing prior to commitment to production hardware. Subassembly testing or testing only in the plant would involve schedule and cost risk if equipment does not function as expected.

4. SCHEDULE REQUIREMENT

Completion of prototype testing prior to release of hardware production drawings (9/93).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Fallback position is alternative (2.1) above. If prototype testing is not performed, unacceptable performance of production hardware in hot flow tests in the plant will cause cost and schedule risks.

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Originator Date

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G. C. Bramblett 3.25.87
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DATE: 2/27/87

MAIN CIRCULATOR MOTOR COOLING DESIGN VERIFICATION
DDN M.21.01.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Submerged motor and bearing cooling is produced by shaft mounted fans circulating helium through cooling passages into a collection cavity that feeds the cooling helium flow into the water cooled heat exchanger. Distribution of internal cooling flow through the motor, bearings, rotor, stator and windings needs validation.

1.1 Summary of Function Number/Title/Assumptions

F1.1.2.2.1.2.2.1 "Cool Motor," Assumption 1: Adequacy of motor and bearing cooling flow can be established by detailed analysis and validated by experimental data.

1.2 Current Data Base Summary

No experimental data is available for cooling of submerged motor configuration in helium.

1.3 Data Needed

Data on cooling flows, pressures and temperatures in the motor cavity are required to validate the thermal/hydraulic performance of the submerged motor cooling system.

Quality assurance must satisfy QAL II requirements.

1.4 Data Parameters/Service Conditions

The following parameters for a full scale submerged motor will provide data sufficient for extrapolation to design conditions.

Pressure (psia)	Ambient - [18]
Temperature (°F)	Ambient - [300]
Cooling Tube Temperature (°F)	[50-80]
Cooling Water Pressure (psia)	[50-100]
Water Flow (gpm)	[50-100]
Drive Power (hp)	[50]
Heat Load (kW)	[30]

2. DESIGNER'S ALTERNATIVES

The following alternative is available:

2.1 Rely on analysis and confirm analytical predictions in main circulator prototype test (M.21.01.02).

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to obtain experimental data on motor cooling flow conditions in air early in the design process. The alternative of obtaining confirmatory data in the prototype test will increase schedule risk if motor cooling performance is not as predicted.

4. SCHEDULE REQUIREMENTS

Validated motor cooling capability six months prior to start of main circulator prototype manufacture (9/91).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Validate submerged motor cooling performance in main circulator prototype testing (2.1). Failure to confirm analysis by component test could result in schedule delays and/or cost impact.

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Originator Date

R. F. Turner 2/27/87
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concurrence for
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DETERMINE PROPERTIES OF 2-1/4 Cr - 1 Mo BASE METAL AND WELDMENTS DDN M.21.02.02 PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insufficient property data base exists for the design of steam generator tubing and plates fabricated from 2 1/4 Cr - 1 Mo material, with and without weldments, which operate in elevated temperature regimes.

1.1. Summary of Function/Title/Assumptions

F1.1.2.3.1, "Transfer Heat from Primary Coolant to Heat Exchanger", Assumption 8: Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metals used in steam generator.

F1.1.2.3.2, "Transfer Heat from Exchanger to Secondary Coolant", Assumption 11: Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metals used in steam generator.

F2.1.2.3.1, "Protect the Capability to Transfer Heat from Primary Coolant to Heat Exchanger", Assumption 4: Proposed new creep-fatigue damage rules to Code Case N-47 for 2-1/4 Cr - 1 Mo and Alloy 800H are conservative, Assumption 5: Changes to Code Case N47 concerning weldment properties of 2-1/4 Cr - 1 Mo and Alloy 800H will not significantly affect the steam generator design.

F2.1.2.3.2, "Protect the Capability to Transfer Heat from Heat Exchanger to Secondary Coolant", Assumption 4: Proposed new creep-fatigue damage rules to Code Case N-47 for 2-1/4 Cr - 1 Mo and Alloy 800H are conservative, Assumption 5: Changes to Code Case N47 concerning weldment properties of 2-1/4 Cr-1 Mo and Alloy 800H will not significantly affect the steam generator design.

1.2. Current Data Base Summary

The primary coolant contains impurities which can cause corrosion in the form of oxidation, decarburization and carburization. At the design temperatures of the components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep fatigue and crack growth. In the tests performed to date, decarburization while exposed to sodium caused significant changes in the tensile and creep properties. Under most conditions the property changes involved a reduction in strength. Also, from the creep-rupture tests in HTGR helium environment it was observed that 2-1/4 Cr - 1 Mo steel is weaker than predicted from vendor air data. Available information on the creep-fatigue behavior of 2-1/4 Cr - 1 Mo steel have indicated that this material tested in helium environment has improved fatigue life for all test wave forms in comparison with tests in air except for the tensile hold only tests. Some limited test data obtained recently on 2-1/4 Cr - 1 Mo weldments suggests that these weldments might be as strong as the base metal.

Limited data to date has showed that for this material, the thermal aging while exposed to argon caused significant reduction in the tensile and creep properties.

Recently, ASME Code Case N47 came out with a creep-fatigue design curve for 2-1/4 Cr - 1 Mo that represents an approach of using total damage (D) values which intersect at values of 0.1 and 0.1 for all temperatures. There were other suggested approaches to the creep-fatigue design curves for this material; however, this approach, which is based on data taken in air, was adopted for the time being.

There are some fracture mechanics data for annealed 2-1/4 Cr - 1 Mo. These include fracture toughness data and fatigue crack growth data. The fatigue crack growth data show an increase in crack growth rate relative to the lower test temperature results. There are very little creep crack growth rate data available.

1.3. Data Needed

Data are needed to determine the effects of exposure to primary coolant (He) with its design impurities and temperature over design life on selected properties of 2-1/4 Cr - 1 Mo base metal and its weldments. These data need to be sufficient to quantify to a 95% confidence that these properties meet or exceed design values. Quality Assurance must satisfy the requirements of QAL-II.

1.4. Data Parameters/Service Conditions

Sufficient data are required to determine the effects of thermal aging and corrosion at temperatures from 700^{o}F (371^{o}C) to 1100^{o}F (593^{o}C) for up to 3 X 10^5 hours on the following properties of 2-1/4 Cr - 1 Mo tubing and plate material and its weldments. The properties should be obtained in helium primary coolant with impurities up to 2ppmv H_2O, 7ppmv $CO+CO_2$, 10ppmv H_2, 2ppmv CH_4, and 10ppmv N_2.

a. Tensile properties for tubing and plate material in helium as a function of temperatures from 400^{o}F (204^{o}C) to 1100^{o}F (593^{o}C) with design values for:

- Ultimate tensile strength (S_u) of not less than 0.8 S_u for as received material.

- Yield strength (S_y) of not less than 0.8 S_y for as received material.

- Elastic Modulus (E) within $\pm$ 20% of E for as received material.

- Stress-Strain values up to 2% total strain.

b. Cyclic stress/strain properties in helium for tubing and plate material at temperatures from 400°F (204°C) to 1100°F (593°C) for at least first 10 cycles.

c. Low cycle fatigue strain range ($\Delta\varepsilon$) to 1,000 cycles for tubing and plate material in helium at temperatures from 400°F (204°C) to 1100°F (593°C) with a design value of at least 5 X 10^{-4} in/in.

d. Creep properties for tubing and plate material in helium at temperatures from 700°F (371°C) to 1100°F (593°C) for duration up to 3 X 10^5 hours with design values for:

  - Minimum stress to rupture (S_r) of not less than S_r for as received material.

  - Stress to cause 1% total strain (S_1) of not less than S_1 for as received material.

  - Time to onset of tertiary creep.

e. Creep-fatigue interaction properties for tubing and plate material in helium at temperatures from 700°F (371°C) to 1100°F (593°C) for 400 cycles with a strain range of at least .05% and hold times of 750 hours to represent steam generator operating cycles.

f. Fracture toughness properties for the tubing material in helium at temperatures from 400°F (204°C) to 1000°F (538°C) with design values for:

  - Critical stress intensity factor (K_{IC}) of greater than 45 ksi $\sqrt{\text{in}}$ (49.5 MPa $\sqrt{\text{m}}$).

  - Fatigue crack growth rate (da/dn) of less than 3.25 X 10^{-5} in/cycle (8.25 X 10^{-4} mm/cycle) for $\Delta K \leq$ 41 ksi $\sqrt{\text{in}}$ (45 MPa $\sqrt{\text{m}}$) and R = 0.0 to 0.75.

  - Creep crack growth rate (da/dt) of less than (TBD in/s) for K (TBD ksi $\sqrt{\text{in}}$).

  - Effect of hold times on crack growth rates.

2. DESIGNER'S ALTERNATIVES

The alternatives are as follows:

2.1. Redesign to reduce stress and accommodate additional allowance for changes in material properties. Relocate, strengthen or insulate critical weldments.

2.2. Reduce helium temperature to limit environmental effects.

2.3. Modify helium gas chemistry to reduce environmental effects.

2.4. Use alternate base metal and weldments materials.

2.5. Design for significantly enhanced inspectability or replacability.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to use 2-1/4 Cr - 1 Mo material for the economizer/evaporator/superheater tubes, tube support plates, tube bundle supports and the shrouds and to generate experimental data to quantify the effects of the primary coolant chemistry and temperature on the properties of this material.

Alternative 2.1 would make the design more expensive and could involve the risk of licensing delays as one tries to justify that the additional allowances are adequate. Alternatives 2.2 and 2.3 would have major impact on plant economics. Alternative 2.4 involves uncertainties which could require DDNs to resolve which could be even more expensive. Alternative 2.5 will increase the cost of the components significantly and for some components may be very difficult if not impossible to satisfy.

It is judged that performing the tests and obtaining the material properties is the most practical and economic solution.

4. SCHEDULE REQUIREMENTS

Interim results, as they become available, on all data defined in Section 4.3.4 are needed throughout the conceptual and preliminary design phases (10/89). Final results on all data are needed 12 months into the final design phase for incorporation into the final design and for ASME Code action. This implies that all data are needed by 10/90.

5. PRIORITY

Urgency: 1
Cost Benefit: H
Uncertainty in Existing Data: H
Importance of New Data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NON-EXECUTION

The fallback position is Alternative 2.1. The consequences to the program of non-execution would involve design modifications, higher cost, risk of customer and licensing nonacceptance, or significantly enhanced NDE and ISI requirements enforced by NRC. The ultimate results could be the risk of plant operation being limited as data is generated by others.

M. Basol 2/23/87
Originator Date

[illegible] 2/26/87
Department Manager Date

[illegible] 2/27/87
Manager, Project Operations Date

DETERMINE PROPERTIES OF
ALLOY 800H BASE METAL
AND WELDMENTS
DDN M.21.02.03
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) Modular HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

An insufficient property data base exists for the design of steam generator tubing, plates and forging fabricated from Alloy 800H material, with and without weldments, which operate in elevated temperature regimes.

1.1. Summary of Function/Title/Assumptions

F1.1.2.3.1, "Transfer Heat from Primary Coolant to Heat Exchanger, Assumption 8: Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metals used in steam generator.

F1.1.2.3.2, "Transfer Heat from Exchanger to Secondary Coolant", Assumption 11: Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metals used in steam generator.

F2.1.2.3.1, "Protect the Capability to Transfer Heat from Primary Coolant to Heat Exchanger", Assumption 4: Proposed new creep-fatigue damage rules to Code Case N47 for 2-1/4 Cr 1 Mo and Alloy 800H are conservative, Assumption 5: Changes to Code Case N47 concerning weldment properties of 2-1/4 Cr - 1 Mo and Alloy 800H will not significantly affect the steam generator design.

F2.1.2.3.2, "Protect the Capability to Transfer Heat from Heat Exchanger to Secondary Coolant", Assumption 4: Proposed new creep fatigue damage rules to Code Case N47 for 2-1/4 Cr - 1 Mo and Alloy 800H are conservative, Assumption 5: Changes to Code Case N47 concerning weldment properties of 2-1/4 Cr - 1 Mo and Alloy 800H will not significantly affect the steam generator design.

1.2. Current Data Base Summary

The primary coolant contains impurities which can cause corrosion in the form of oxidation, decarburization and carburization. At the design temperatures of Alloy 800H components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation, along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep-fatigue, and crack growth. In the creep-rupture tests performed to date, the specimens which have ruptured exhibited rupture times of 1,000 - 32,000 hours, which fall within the air data scatter bands. These results, however, should not be interpreted to imply that there are no environmental effects on the creep-rupture properties of Alloy 800H because these data represent tests of fairly short duration when compared to the design life of 300,000 hours.

The data available at thermal aging times up to 20kh indicate that at elevated temperatures of 538°C to 593°C (1000°F to 1100°F), the yield and ultimate tensile strengths of this material increases slightly with some reduction in ductility. However, at test temperatures above 649°C (1200°F), this trend is reversed.

The test results obtained to date on creep-fatigue interaction indicate that the Linear Damage Design Criterion in the Code Case N47 is highly nonconservative for Alloy 800H at 650°C (1200°F) in air when applied to constant strain hold creep-fatigue loading cycles. A new creep-fatigue damage rule has been proposed as a result.

The weldments at elevated temperatures have been an area of concern because of the potential for limited ductility of weld metal and the potential for high strain concentrations (both metallurgical and geometric) in the heat-affected zone of weldments. Limited test data obtained to date at GA suggests that Inconel 82 clad Alloy 800H plate and Inconel 82 welded Alloy 800H tubes are as strong or stronger than the base metal in both tensile and creep-rupture tests.

Data available to date have shown that the presence of the HTGR helium environment had no discernable effect on the stress behavior during low-cycle fatigue or creep-fatigue testing; however, it did substantially increase the low-cycle fatigue life at 650°C (1202°F relative to air data, due to the reduction in the crack propagation rate.

Fracture toughness data available to date indicate that the room temperature J_{IC}, for material aged 10,000h at 593°C (1100°F) in air, is significantly lower than that measured on material aged for the same length of time at the other temperatures.

Some fracture mechanics data is available on cold worked Alloy 800H at 677°C (1250°F) and 732°C (1350°F) relative to fatigue crack growth and creep crack growth.

1.3. Data Needed

Data are needed to determine the effects of exposure to primary coolant (He) with its design impurities and temperature over design life on selected properties of Alloy 800H base metal and its weldments. These data need to be sufficient to quantify to 95% confidence that these properties meet or exceed design values. Quality Assurance must satisfy the requirements of QAL-II.

1.4. Data Parameters/Service Conditions

Sufficient data are required to determine the effects of thermal aging and corrosion at temperatures from 800°F (427°C) to 1200°F (649°C) for up to 3 X 10^5 hours on the following properties of Alloy 800 tubing, plate and forging material and its weldments. The properties should be obtained in helium primary coolant with impurities up to 2ppmv H_2O, 7ppmv CO + CO_2, 2ppmv H_2, 2ppmv CH_4, and 10ppmvN_2.

a. Tensile properties for tubing, plate and forging material in helium as a function of temperatures from 600°F (316°C) to 1200°F (649°C) with design values for:

- Ultimate tensile strength (S_u) of not less than 0.8 S_u for as received material.

- Yield strength (S_y) of not less than 0.8 S_y for as received material.

- Elastic Modulus (E) within $\pm$ 20% of E for as received material.

- Stress-strain values up to 2% total strain.

b. Cyclic stress/strain properties in helium for tubing, plate, and forging material at temperatures from 600°F (316°C) to 1200°F (649°C) for at least first 10 cycles.

c. Low cycle fatigue strain range ($\Delta\varepsilon$) to 1,000 cycles for tubing, plate and forging material in helium at temperatures from 600°F (316°C) to 1200°F (649°C) with a design value of at least 5 X 10^{-4} in/in.

d. High cycle fatigue strength to $>$1 X 10^6 cycles for tubing material in helium at temperatures from 880°F (471°C) to 1188°F (642°C) with a design value of at least [1.0 ksi].

e. Creep properties for tubing, plate, and forging material in helium at temperatures from 800°F (427°C) to 1200°F (649°C) for duration up to 3 X 10^5 hours with design values for:

- Minimum stress to rupture (S_r) of not less than S_r for as received material.

- Stress to cause 1% total strain (S_1) of not less than S_1 for as received material.

- Time to onset of tertiary creep.

f. Creep-fatigue interaction properties for tubing, plate and forging material in helium at temperatures from 800°F (427°C) to 1200°F (649°C) for 400 cycles with a strain range of at least .05% and hold times of 750 hours to represent steam generator operating cycles.

g. Fracture toughness properties for the tubing and forging material in helium at temperatures from 600°F (316°C) to 1200°F (649°C) with design values for:

- Critical stress intensity factor (K_{IC}) of greater than 70 ksi $\sqrt{in}$ (77 MPa $\sqrt{M}$).

- Fatigue crack growth rate (da/dn) of less than 3.25×10^{-5} in/cycle (8.25×10^{-4} mm/cycle) $\Delta K \leq 43$ ksi $\sqrt{in}$ (47 MPa$\sqrt{m}$) and R = 0.0 to 0.75.

- Creep crack growth rate (da/dt) of less than (TBD in/s) for K of (TBD ksi $\sqrt{in}$).

- Effect of hold times on crack growth rates.

2. DESIGNER'S ALTERNATIVES

The alternatives are as follows:

2.1. Extrapolate existing creep-rupture data to full design life.

2.2. Design for significantly enhanced inspectability or replaceability.

2.3. Reduce helium temperature to limit environmental effects.

2.4. Modify helium gas chemistry to reduce environmental effects.

2.5. Redesign to reduce stress and accommodate additional allowance for changes in material properties. Relocate, strengthen or insulate critical weldments.

2.6. Use alternate base metal and weldments materials.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to use Alloy 800H material for the superheated steam tubesheet, finishing superheater tubes, and tube support plates and to generate experimental data to quantify the effects of the primary coolant chemistry and temperature on the properties of this material.

Alternative 2.1 will involve risking of industry acceptance of this inadequate technical base. Alternative 2.2 will increase the cost of the components significantly and for some components may be very difficult if not impossible to satisfy. Alternatives 2.3 and 2.4 would have a major impact on plant economics. Alternative 2.5 would make the design more expensive and could involve the risk of licensing delays as one tries to justify that the additional allowances are adequate. Alternative 2.6 involves uncertainties which could require DDNs to resolve which could be even more expensive.

It is judged that performing the tests and obtaining the material properties is the most practical and economic solution.

4. SCHEDULE REQUIREMENTS

Interim results, as they become available, on all data defined in Section 4.3.4 are needed throughout the conceptual and preliminary design phases (10/89). Final results on all data are needed 12 months into the final design phase for incorporation into the final design and for ASME Code action. This implies that all data are needed by 10/90.

5. PRIORITY

Urgency: 1
Cost Benefit: H
Uncertainty in Existing Data: H
Importance of New Data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NON-EXECUTION

The fallback position is Alternative 2.5. The consequences to the program of non-execution would involve design modifications, higher cost, risk of customer and licensing nonacceptance, or significantly enhanced NDE and ISI requirements enforced by NRC. The ultimate result could be the risk of plant operation being limited as data is generated by others.

M. Basol 2/23/87
Originator Date

2/26/87
Department Manager Date

2/27/87
Manager, Project Operations Date

TUBE BUNDLE ACOUSTIC TEST
DDN M.21.02.04
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) MODULAR HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Acoustic vibrations generated by vortex shedding and flow separation from multiple tubes can be amplified by tuned resonant chambers. The high noise levels produced can cause damage to certain shroud or thermal barrier surfaces.

1.1. Summary of Function/Title/Assumptions

F.1.1.2.2.2.1.1, "Maintain Primary Coolant Boundary Integrity", Assumption 4: Methods will be developed for the timely prediction of acoustic and flow induced loads within the primary coolant loop.

F.1.1.2.3.1.1.1, "Receive Primary Coolant", Assumption 2: Steam generator tube bundle/cavity will not generate excessive acoustic loads on the shrouds.

F.1.1.2.3.1.1.2, "Channel Primary Coolant through Heat Exchanger", Assumption 5: Steam generator tube bundle/cavity will not generate excessive acoustic loads on the shrouds.

F.1.1.2.3.1.1.3, "Discharge Primary Coolant", Assumption 3: Steam generator tube bundle/cavity will not generate excessive acoustic loads on the shrouds.

1.2. Current Data Base Summary

There is considerable amount of literature available that describes the acoustic characteristics of heat exchanger bundles in cross flow and methods to investigate various aspects of heat exchanger acoustic problems.

The only acoustic vibration test results available for a tube bundle geometry similar to current design configuration were generated by Sulzer Brothers, Inc. for FSV steam generators. The conclusions of this testing were that the various acoustic vibrations occurred at or near the frequencies calculated, and that no distinct resonances with the vortex shedding frequencies were found, this being a major advantage of helical bundles over straight tube bundles.

1.3. Data Needed

Data is needed that will produce representative frequency spectra and sound pressure levels generated by the tube bundle as a function of flow velocities and geometry variations. This information is needed to verify lack of destructive acoustic energy obviating need for geometry modifications or baffling. The data generated should satisfy Quality Assurance Level II requirements.

1.4. Data Parameters/Service Conditions

The critical test parameters are geometric similitude, Reynolds number and speed of sound. For model testing in air, where the speed of sound is approximately one-quarter the speed of sound in helium, the model must be one-quarter scale to match the sound wave velocity to the physical dimensions of the model.

The helium gas flow conditions that should be simulated are as follows:

- Temperatures from 500^{o}F to 1300^{o}F
- Pressure: 918 psia.
- Gap velocities from 25 ft/sec. to 42 ft/sec.
- Reynolds Number from 20000 to 31000.
- Speed of sound of 4450 ft/sec. to 6000 ft/sec.

2. DESIGNER'S ALTERNATIVES

2.1. The following alternative is available: Rely on analysis and avoid large resonant plates or other acoustic sensitive surfaces in the design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach involves wind tunnel testing of a 1/4 scale plastic and metal model of the helical tube bundle. The model will be designed so that the bundle geometry can be varied. The sound produced by air flow through the test model as a function of bundle geometry and flow velocity will be measured. The task includes generating the test specification, fabricating the model, testing, and documentation of the results.

The steam generator design incorporates shrouds, flow baffles and/or shields which may be acoustically sensitive. Theoretical analysis by itself is not considered adequate. (Alternative 2.1).

4. SCHEDULE REQUIREMENTS

Data is needed by start of the Final Design phase (10/89).

5. PRIORITY

Urgency: 2
Cost Benefit: M
Uncertainty in Existing Data: M
Importance of New Data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NON-EXECUTION

The fallback position is Alternative 2.1. The consequences to the program of non-execution would involve design modifications and higher cost.

Should additional modifications be required but not be made due to lack of test data, structural damage within the unit could occur during operation or extensive steam generator modifications will be made after initial plant tests.

M. Basol 2/23/87
Originator Date

2/26/87
Department Manager Date

2/27/87
Manager, Project Operation Date

DATE: 2/23/87

LARGE HELICAL COIL PROGRAM
DDN M.21.02.07
PROJECT NUMBER 6300

Plant: 4 X 350 MW(t) Modular HTGR/System 21

1. Requirement or Design Feature Requiring Experimental Data or Validation Testing

Determine the feasibility of coiling and threading multiple tubes in concentric coils through holes in full and partial support plates. Concerns are: clearances between tube and plate; wear protection device installation; tolerances; and fabrication time.

1.1 Summary of Function/Title/Assumptions

F1.1.2.3.1.2, "Support Heat Transfer Surfaces", Assumption 3: The method of manufacturing of a helical coil bundle with drilled radial tube support plates will be verified.

1.2 Current Data Base Summary

Several full-scale fabrication tests and production bundles have demonstrated the feasibility of coiling and threading tubes through similar support structures. However, the feasibility has been demonstrated on tube bundles of smaller diameters, fewer number of coils, shorter tube lengths, and fewer number of tubes. The differences in these parameters, in addition to the differences in the details of the support structure, create a concern over the applicability of the current data base to the Modular HTGR Steam Generator design. Aside from the differences, the data base would provide useful information to establish a more productive and efficient test program.

1.3 Data Needed

The data needed from the performance of this test is to confirm that the fabrication of a large helical bundle is feasible and can be accomplished in a reasonable timeframe. This data shall satisfy Quality Assurance Level II requirements.

1.4 Data Parameters/Service Conditions

Data parameters needed from the test shall include:

1.4.1 detailed fabrication procedures for handling, coiling, and threading tubes (i.e., quantity and placement of support points, bending rate, thread-in forces, etc.);

1.4.2 types and designs of tooling required for performing operations (i.e., tube support rolls, tube wear protection device upsetting tool, thread-in tools, etc.);

1.4.3 tolerances (tube forming; support plate holes, etc.);

1.4.4 fabrication time (broken down for each operation).

2. Designer's Alternatives

Available alternatives are:

2.1 Different helical bundle support system that would not require fabrication testing;

2.2 Different tube bundle geometry and configuration that would not require fabrication testing;

2.3 Adapt data base parameters and information to current design;

2.4 Arbitrarily increase fabrication time to allow for unknowns and potential problem areas.

3. Selected Design Approach and Explanation

The selected approach involves coiling and threading a number of selected tubes into a full- scale, drilled plate support structure. Tubes will be selected to fully represent the gamut of coiling and threading possibilities.

This selection is based upon utilization of the helical bundle supported by solid, drilled plates. The helical bundle is the most compact heat exchanger design for this application and the solid, drilled plate support system appears to best satisfy the requirements of thermal expansion and seismic load paths. Adaption of the data base poses questions and is felt non-applicable because of the many differences in bundle parameters. Arbitrarily increasing the fabrication schedule to allow for unknowns and learning would directly and adversely affect the cost as well as the "real" schedule.

4. Schedule Requirements

Data is needed prior to the start of the Final Design Phase (10/89).

5. Priority

Urgency: 2
Cost Benefit: H
Uncertainty on Existing Data: H
Importance of New Data: H

6. Fallback Position and Consequences of Non-Execution

The first-level fallback position should only consider an adaptation of the current data base or arbitrarily increasing the fabrication span. The next fallback position is to investigate different support system designs that would ease manufacturing and not require preliminary testing. The last fallback position is to investigate alternate tube bundle geometry.

The consequences of non-execution of this test will leave a concern of the fabrication feasibility of a large helical coil bundle. Proceeding with the current design but without fabrication testing would result in increased costs and increased fabrication schedule.

R. W. Fulkhoute 2/24/87
Originator Date

Alan Wh 2/26/87
Department Manager Date

A. McEwliste 2/27/87
Mgr., Project Operation Date

AIR FLOW TEST
DDN M.21.02.08
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) MODULAR HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The primary coolant flow entering the FSH tube bundle must satisfy certain constraints relative to the magnitude and location of hot/cold streaks and velocity maldistribution. The presence of hot/cold streaks will cause temperature stratification which may be necessary to reduce through mixing in order to satisfy BMW temperature constraints. This flow distribution problem may be further complicated by the fact that the cross duct exit (after the 90° elbow) and the FSH tube bundle may not be coaxial.

1.1. Summary of Function/Title/Assumptions

F1.1.2.3.1.1.2, "Channel Primary Coolant through Heat Exchanger", Assumption 1: Peak/average helium velocity ratio is 1.10 at entrance to the FSH tube bundle.

F2.1.2.3.2, "Protect the Capability to Transfer Heat from Heat Exchanger to Secondary Coolant", Assumption 1: The BMW must not be wetted; Assumption 2: 50°F superheat is acceptable for BMW; Assumption 3: The BMW tube wall temperature less than or equal to 900°F is acceptable.

1.2. Current Data Base Summary

Analytical methods such as flow distribution codes are available for predicting the flow field that can be realized for a given geometric configuration and inlet conditions. Experimental data exists on the hydraulic resistance of screens and baffles.

1.3. Data Needed

The velocity and temperature profiles of the primary coolant flow as it enters the FSH tube bundle. The approximate elevation in the FSH tube bundle where the velocity profile becomes uniform. The stagnation pressure losses from the cross duct exit (after the 90° elbow) to the entrance of the FSH tube bundle. Quality Assurance must satisfy the requirements of QAL II. The above data will be generated in conjunction with DDN M.21.00.01.

1.4. Data Parameters/Service Conditions

The velocity and temperature profiles and the stagnation pressure losses for a cross duct Reynolds number range of from 1.25×10^6 (25% power) to 3.65×10^6 (100% power).

a. Circumferential and radial velocity measurements at:

- The inlet to the eccentric duct.
- The outlet of the eccentric duct.
- The inlet to the FSH tube bundle.
- The lower elevation of the FSH.

b. Circumferential and radial static temperature measurements at:

- The inlet to the eccentric duct.
- The outlet of the eccentric duct.
- The inlet to the FSH tube bundle.
- The lower elevation of the FSH.

c. Circumferential and radial static and stagnation pressure measurements at:

- The cross duct upstream of the 90° elbow.
- The inlet to the eccentric duct.
- The outlet of the eccentric duct.
- The inlet to the FSH tube bundle.

2. DESIGNER'S ALTERNATIVES

The alternative is as follows:

2.1. Rely solely on analytical methods (flow distribution codes) and utilize the available experimental data on the hydraulic resistance of screens, baffles, and gas mixing devices.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to perform the air flow test to determine the primary coolant velocity and temperature profiles and the location of flow distribution devices which will yield the required radial and circumferential velocity and temperature profiles.

Alternative 2.1 involves uncertainty in meeting the BMW temperature constraints and uncertainty in determining the tubeside orifice requirements which, in turn, translates into uncertainty in specifying the feedwater pressure.

It is judged that performing the test is imperative to support the plant design.

4. SCHEDULE REQUIREMENTS

Test data are needed during the Preliminary Design Phase to remove uncertainties in the design (1988).

5. PRIORITY

Urgency: 1
Cost Benefit: H
Uncertainty: H
Importance: H

6. FALLBACK POSITION AND CONSEQUENCES OF NON-EXECUTION

The fallback position is Alternative 2.1. The consequences to the program of non-execution involve plant design modifications and risk of customer and licensing nonacceptance. Plant operation and plant life would be greatly limited.

R. W. Hanselay 3-16-87
Originator Date

A. W. Lubkowitz for A. H. Sprung 3-18-87
Department Manager Date

A. W. Lubkowitz for A. D. McWhorter 3-18-87
Manager, Project Operation Date

STEAM GENERATOR INSULATION
VERIFICATION TESTS
DDN M.21.02.10
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) MODULAR HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Thermal and mechanical performance is of concern due to the unique configuration which requires blind installation and, therefore, enhances the concern of insulation accessibility for maintenance or alteration once the steam generator is installed.

1.1. Summary of Function/Title/Assumptions

F1.1.2.3.1.1, "Flow Primary Coolant Through Heat Exchanger", Assumption 3: Insulation required at steam generator inlet plenum and outer shroud.

1.2. Current Data Base Summary

Considerable amount of literature is available relative to high temperature insulation physical and thermophysical properties. Variety of insulations are available in special forms to meet specific service requirements.

1.3. Data Needed

Physical and operational characteristics of insulation are required. Specific data needed would be relative to thermal cycling of fibrous insulation, mechanical and acoustic vibrations, effects of flow and thermal gradients. These tests will produce temperature data for certain critical components of steam generator and verify proposed thermal barrier for the life of the plant. Additional test data relative to any destructive impact on insulation due to vibrations and sliding contacting surfaces would be obtained. Quality Assurance must satisfy the requirements of QAL II.

1.4. Data Parameters/Service Conditions

Sufficient test data is required to demonstrate the adequacy of the installed insulation under simulated critical environmental conditions. The insulation materials to be tested consist of (TBD). These will be subjected to flow velocities, thermal cycling, mechanical and acoustic vibrations, and sliding loads depending on the critical locations of interest. The test data will be used to confirm the thermal and mechanical viability of the chosen insulation materials and the methods of installation.

The service conditions are as follows:

- Normal operating conditions:

  - $500^{o}F$ to $1300^{o}F$ temperature range.
  - Helium environment.
  - Duration 3×10^{5} hours.
  - Vibration frequency range of (TBD).
  - Acoustic sound pressure levels of (TBD).
  - Sliding contact forces of (TBD).

  - Flow velocity range of (TBD).

- Off-normal operating conditions:

  - TBD.

2. DESIGNER'S ALTERNATIVES

2.1. Thermal analyses to determine the thermal performance adequacy of the insulation.

2.2. Rely on manufacturer insulation specification for mechanical performance.

3. SELECTED APPROACH AND EXPLANATION

The selected approach is to perform testing of different critical regions under simulated environment conditions. Thermal performance of the insulation can be obtained by analysis, however, analysis alone is not sufficient to assure the mechanical performance of the insulation.

Performing the described tests is the only way of checking the mechanical performance of the insulation.

4. SCHEDULE REQUIREMENTS

Data is needed one year into the final design phase (10/90).

5. PRIORITY

Urgency: 3
Cost Benefit: M
Uncertainty in Existing Data: M
Importance of New Data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NON-EXECUTION

The fallback positions would be to rely on alternates 2.1 and 2.2. The insulation system will have to be redesigned to avoid areas of high vibration and acoustic loads. The consequences of these on the program would be higher cost of redesign and possibility of failure of critical steam generator component and/or higher heat loss due to dislocation resulting in lower thermal performance of insulation.

Naeem U. Karim 2-26-87
Originator Date

Alan W. [illegible] 2/26/87
Department Manager Date

Alan M. [illegible] 2/27/87
Manager, Project Operation Date

VIBRATIONAL FRETTING WEAR AND SLIDING WEAR PROTECTION TESTS
DDN M.21.02.11
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) MODULAR-HTGR/System 21

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Of concern is the wear protection method for the steam generator tubes which are in direct contact with the drilled support plates and other areas of the design that might require metal to metal contact.

1.1. Summary of Function/Title/Assumptions

F2.1.2.3.1.2, Protect the Capability to Support Heat Transfer Surfaces (Assumption 1). Vibrational wear and sliding wear protection methods will be verified.

1.2. Current Data Base Summary

An experimental and analytical study was made at GA Technologies of wear induced by vibration of Alloy 800H and 2-1/4 Cr - 1 Mo heat exchanger tubes in loosely held supports in a helium environment at temperatures up to 650°C (1200°F). Some of the findings of this program were: Impact wear can be minimized by reducing impact contact stresses below the fatigue allowable stress; the maximum impact velocity is limited by the tube-support clearance and the peak-to-peak midspan vibration velocity; decreasing vibration amplitude or tube-support clearance markedly decreases wear rate and surface deformation at all temperatures; application of plasma-sprayed chromium carbide coating to both the tube and the plate specimens greatly reduces the surface self-adhesion and surface deformation which would otherwise occur above 500°C (930°F). Below this temperature, a coating is probably not required.

Sliding wear and spallation tests conducted at GA Technologies indicate that both 2-1/4 Cr - 1 Mo and Alloy 800H alloys show severe wear under simulated HTGR conditions. The wear mechanism in most cases for both alloys was due adhesion dominated. It is concluded that a definite need for protective coating exists. Chromium carbide-nichrome coated Alloy 800H wear test specimens showed wear, but no evidence of adhesion, a significant improvement over the results on uncoated Alloy 800H wear test specimens.

1.3. Data Needed

At the tube support interfaces several different forms of wear could take place. In the HTGR environment, this could be in the form of fretting wear due to flow-induced vibration, sliding wear and galling due to axial sliding motion or circular sliding motion and impact wear due to tube bouncing inside the support plates. These forms of wear could be protected by the use of a wear protection device or by coating the metal surfaces. Wear data (wear rates, friction coefficient, etc.) are needed under representative service conditions and materials, to finalize wear protection methodology, including a decision on which components of the steam generator need coating, and a selection of a coating material. Also needed are data to determine the coating thicknesses required for a 40-year plant life and whether the coatings need to be ground smooth prior to use to minimize friction and wear rates. Data satisfying Quality Assurance Level II requirements are needed.

1.4. Data Parameters/Service Conditions

Vibrational fretting wear data is needed in dry helium up to the high helium temperature on Alloy 800H and 2-1/4 Cr - 1 Mo tubing material with identical support materials.

Wear rates and wear characteristics (i.e., wear depth, wear volume and/or weight loss, friction coefficient, etc.) as a function of:

- Temperature,
- Vibration amplitude,
- Clearance between tube O.D. and support I.D.,
- Number of cycles,
- Static Load,
- Frequency,
- With and without wear coating,
- Surface conditions.

In addition to above, the wear coating stability under the service conditions of interest and the possible effects of the wear coating material and method of application on the base material properties needs to be investigated.

Previous sliding wear data should be expanded to determine the coating thicknesses required for the design life of the plant and to determine whether the coatings need to be ground smooth prior to use to minimize friction and wear rates.

The service conditions are as follows:

1. Normal Operating Conditions:

   - 400^oF (204^oC) to 1300^oF 704^oC) temperature range

   - Helium primary coolant with impurities up to 2ppmv H_2O, 7ppmv CO + CO_2, 2ppmv CH_4, and 10ppmv N_2, 10ppmv H_2.

   - Duration 3 X 10^5 hours

   - Cycles > 1 X 10^6

   - Vibration frequency range 20 H_z - 200 H_z

   - Static Load - TBD

   - Vibration Amplitude - TBD

2. Off-Normal Operating Conditions:

   - To be determined.

2. DESIGNER'S ALTERNATIVES

The following alternative is available:

2.1. Utilize previous vibrational fretting wear and sliding wear data and laboratory tests of wear coatings.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to perform vibrational fretting wear and sliding wear protection tests. Previous vibrational fretting wear and sliding wear data are limited and focused on different design characteristics. By performing the tests under actual service conditions and current configuration, it is expected that data will be obtained from these tests to provide the data base for the development of wear protection device and/or method in the helical tube bundles.

Alternative 2.1 data relative to vibrational fretting wear do not represent the current steam generator helium primary coolant impurity levels specified.

4. SCHEDULE REQUIREMENTS

The results to be available prior to the completion of the Preliminary Design phase (9/89).

5. PRIORITY

Urgency: 2
Cost Benefit: M
Uncertainty in Existing Data: M
Importance of New Data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NON-EXECUTION

The fallback position is Alternative 2.1. This would mean using data that is not representative of the specified service conditions.

The consequences of non-execution of this test program would be the risk of steam generator reliability and possible requirement for early replacement of the tube bundle.

M. Basol 2/23/87
Originator Date

2/26/87
Department Manager Date

A. Wrobliski 2/27/87
Manager, Project Operations Date

TUBE WEAR PROTECTION DEVICE TESTS
DDN M.21.02.12
PROJECT NUMBER 6300

PLANT: 4 X 350 MW(t) MODULAR HTGR/System 21

1. REQUIREMENT OF DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The HTGR steam generator tubes are supported by drilled plates (EES and FSH bundles) which are in direct contact with the tubes. Currently a wear protection device design exists, however, in the present design there is serious doubt that this device will be able to perform all of its functions at the hot end of the tube bundle. This coupled with the installation difficulties and the cost of this wear protection device indicates a need for development of a better and simpler design.

1.1. Summary of Function/Title/Assumptions

F2.1.2.3.1.2, Protect the Capability to Support Heat Transfer Surfaces (Assumption 1). Vibrational wear and sliding wear protection methods will be verified.

1.2. Current Data Base Summary

There are several functions of the wear protection device in the current steam generator design. These are: to suffice as a manufacturing shim; to afford a sacrificial wear material; to provide vibration damping as required; and to transmit seismic loads from the tube to the radial support plates. (This last function is being studied by CE at this time.)

A "sleeve and wedge" type wear protection assembly was used in FSV and THTR designs. The main function of the assembly in these designs was to protect against vibration induced tube fretting damage. They were not specifically designed to transmit seismic loads.

1.3. Data Needed

Data are needed to confirm the selected wear protection devices adequacy to perform the functions required during the steam generator operation. Data satisfying Quality Assurance Level II requirements are needed.

1.4. Data Parameters/Service Conditions

Sufficient test data are required to demonstrate that the chosen wear protection device will perform all the required functions throughout the design life of the plant. That is, the device should be shown: to protect the tubes from the vibrational fretting wear; to provide sufficient vibrational damping; plus it should be shown that creep of the rings do not completely relax the pre-load of the device on the tubes. It is also required to demonstrate that the installation of this device under shop conditions will be relatively easy.

The details of the testing conditions will be available upon the completion of the analytical studies at Combustion Engineering.

2. DESIGNER'S ALTERNATIVES

The alternative is as follows:

2.1. Do not test. Rely on analysis and Fort St. Vrain experience for the EES and FSH tube wear protection devices.

2.2. Develop alternate tube bundle support designs that do not require wear protection devices.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to test the wear protection device or devices that have been chosen as a result of engineering trade studies oriented toward improving the design. These tests will build confidence base that these devices will perform the desired functions throughout the life of the plant.

Alternate 2.1 was not chosen because analysis alone cannot adequately address all of the concerns. In addition, the testing of the devices used in Fort S. Vrain steam generators have shown them to relax their pre-load on the tubes after some time.

Alternate 2.2 involves very detailed and costly trade studies, and the alternate tube bundle method support could introduce new concerns into the design.

4. SCHEDULE REQUIREMENTS

The results to be obtained prior to completion of Preliminary Design (9/89).

5. PRIORITY

Urgency: 2
Cost Benefit: M
Uncertainty in Existing Data: H
Importance of New Data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NON-EXECUTION

Fallback position is Alternative 2.1. This could lead to use of a device with derated life that requires periodic replacement. This would be a very costly alternative.

The consequences of non-execution of the test program would be the risk of reduced steam generator reliability.

M. Basol 2/23/87
Originator Date

Alan M Lin 2/26/87
Department Manager Date

G. M Roberts 2/27/87
Manager, Project Operation Date

DATE: 2/25/87

VERIFY NSSS ANALYTICAL INSTRUMENTATION SYSTEM
DDN M.30.01.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 30

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Various low level chemical and radiological impurities in the primary coolant helium need to be monitored by appropriate analytical instrumentation. The purpose of this DDN is to verify that readily available commercial instrumentation can perform the necessary monitoring.

1.1 Summary of Function/Title/Assumptions

F1.1.2.2.2.4 "Monitor Primary Coolant Impurities"/Assumption 1; F1.1.2.2.2.4.1 "Sense Parameters"/Assumption 1; F1.1.4.1.1.2.2.1 "Monitor Circulating and Plateout Activities"/Assumption 1; F1.1.4.1.1.2.2.1.1 "Sense Parameters"/Assumption 1; F2.1.2.2.2.4 "Protect the Capability to Monitor Primary Coolant Impurities"/ Assumption 1; F2.1.2.2.2.4.1 "Protect the Capability to Sense Parameters"/Assumption 1; F2.1.4.1.1.2.2.1 "Protect the Capability to Monitor Circulating and Plateout Activities"/Assumption 1; F2.1.4.1.1.2.2.1.1 "Protect the Capability to Sense Parameters"/ Assumptions; F3.1.1.2.2.4 "Monitor Circulating and Plateout Activities"/Assumption 1; F3.1.1.2.2.4.1 "Sense Parameters"/ Assumption 2: It is assumed that the NSSS Analytical Instrumentation is capable of measuring primary coolant moisture, chemical composition, and radioactive isotope concentration with the required accuracy and availability.

1.2 Current Data Base Summary

The present data base for NSSS analytical instrumentation is based on Fort St. Vrain equipment data. The Fort St. Vrain analytical moisture monitor is nonlinear below 10 ppmv and accurate measurements below 2 ppmv are not possible. The Fort St. Vrain gas chromatograph is not automatic. The Fort St. Vrain analytical instruments use technology which is rapidly becoming obsolete. It is desirable to use modern commercially manufactured analytical instrumentation in the 4 x 350 MW(t) HTGR if available and can be proven to meet design requirements.

1.3 Data Needed

NSSS Analytical Instrumentation moisture monitor calibration curve and response time.

Gas chromatograph calibration.

CO analyzer calibration.

Radioactive noble gas monitor calibration and response time.

Radioactive iodine monitor calibration and response time.

General circulating activity monitor calibration and response time.

Tritium monitor calibration and response time.

Plateout probe calibration.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

NSSS Analytical Instrumentation Data Parameters:

Verification of the NSSS Analytical Instrumentation moisture monitor calibration and response time from [1.0] ppmv moisture to [10,000] ppmv moisture at plant service conditions. Verification of gas chromatograph to discriminate expected primary coolant chemical composition at plant service conditions. Verification of CO analyzer calibration from [1.0] ppmv CO to [5000] ppmv CO. Verification of radioactive noble gas monitor calibration and response time at plant service conditions for Kr-88 from [0.10] Ci/lb-He to [5.00] Ci/lb-He and for Xe-133 from [0.10] Ci/lb-He to [20.0] Ci/lb-He. (Other noble gas isotope concentrations to be specified later). Verification of radioactive iodine monitor calibration and response time at plant service conditions for I-131 from [0.001] Ci/lb-He to [0.15] Ci/lb-He. (Other iodine isotope concentrations to be specified later). Verification of primary coolant general radioactivity monitor calibration and response time up to a total primary coolant radioactivity of [35.0] Ci/lb-He. (Radioactive isotopes to be specified later). Verification of tritium monitor calibration and response time for H-3 from [0.0005] Ci/lb-He to [0.75] Ci/lb-He. Verification of plateout probe calibration for species of Cs, I, and S.

Service Conditions:

Primary Coolant	Helium
Primary Coolant Pressure	
(Design)	[1041] psia
(Operating)	[925] psia

Primary Coolant Temperature	
(Design)	[1300]°F
(Operating)	[1268]°F
Primary Coolant Moisture	1.0 to 10,000 ppmv
Primary Coolant Flow Rate	[1,246,000] lb/h
Primary Coolant Radioactivity: (Design)	Kr-88 [2.0] Ci/lb-He
	Xe-133 [0.87] Ci/lb-He
	I-131 [0.006] Ci/lb-He
	Sr-90 [1.29] µCi/lb-He
	Ag-110m [100.0] µCi/lb-He
	Cs-137 [25.4] µCi/lb-He
	H-3 [700.0] µCi/lb-He
Primary Coolant Chemical Contaminant Concentration:	O_2 [10] ppmv
	CO [5] ppmv
	CO_2 [2] ppmv
	H_2 [10] ppmv
	N_2 [10] ppmv
	H_2S [trace] ppmv
	CH_4 [2] ppmv

2. DESIGNER'S ALTERNATIVES

2.1 Use commercial analytical instrumentation without performing design verification.

3. SELECTED DESIGN APPROACH AND EXPLANATION

It is recommended that one or more commercially available analytical instruments be selected for each of the analytical instrumentation components: (1) moisture monitor, (2) noble gas monitor, (3) iodine monitor, (4) tritium monitor, (5) CO analyzer, (6) general radioactivity monitor, and (7) gas chromatographs. Each candidate component monitor should then be operated under conditions to simulate modular HTGR plant requirements and verify sensitivity, accuracy, response, and repeatability design requirements are satisfied.

Commercially available analytical instrumentation equipment which has clearly been demonstrated by prior vendor testing or operational experience to meet modular HTGR design requirements will be accepted as qualified for use in the modular HTGR plant.

Commercially available equipment may not meet design requirements without some modifications to adapt them to accept hot helium samples. Design verification can prove very valuable in selecting the best candidate instrument to meet sensitivity, accuracy, response, repeatability, and maintainability requirements.

4. SCHEDULE REQUIREMENTS

Design verification of the NSSS analytical instrumentation must be completed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Provide NSSS Analytical Instrumentation without design verification. Failure of any analytical instrumentation equipment to satisfy design requirements would require modifying the design or revising the requirements to accept a lower performance. This would adversely impact both plant schedule and cost.

[signature] 3/12/87
Originator Date

[signature] 031387
Department Manager Date

[signature] for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/25/87

VERIFY PLANT PROTECTION AND INSTRUMENTATION SYSTEM MOISTURE MONITOR
DDN M.32.02.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 32

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Appropriate moisture monitoring is required to detect a leaking steam generator for use in the plant protection system. The purpose of this DDN is to compare candidate sampling methods (rakes) including commercial moisture monitors to confirm necessary performance characteristics.

1.1 Summary of Function/Title/Assumptions

F2.1.1.3.1 "Sense Parameters."/Assumption 1: Moisture monitors adequate to meet accuracy and response time requirements are feasible.

F2.1.1.4.1 "Sense Parameters."/Assumption 1: Moisture monitors adequate to meet accuracy and response time requirements are feasible.

F2.1.4.1.1.2.2.3.1 "Sense Parameters."/Assumption 1: Moisture monitors adequate to meet accuracy and response time requirements are feasible.

F3.1.1.2.2.2.1 "Sense Parameters"/Assumption 1: Moisture monitors adequate to meet accuracy and response time requirements are feasible.

1.2 Current Data Base Summary

The current data base for moisture monitor performance in HTGR applications is based on the operating history of the Fort St. Vrain safety-related dewpoint hygrometers. These hygrometers are a special design which would require a remanufacturing effort for application in new HTGRs. They also require a liquid nitrogen cooling system which has been very troublesome to operate and maintain.

1.3 Data Needed

Plant Protection and Instrumentation System moisture monitor accuracy and time of response at plant service conditions.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

Moisture Monitor Data Parameters

Accuracy of measuring primary coolant helium moisture concentration in the range [800] ppmv to [1200] ppmv at plant service conditions. Accuracy required is [10]% from 1.0 to 10,000 ppmv.

Time of response for measurement of primary coolant moisture concentration from [1] ppmv to [1200] ppmv at plant service conditions. Time constant required is less than [5] s.

Moisture monitor measurement time constant in seconds.

Service Conditions

Primary Coolant	Helium
Primary Coolant Pressure (Design)	[1041] psia
(Operating)	[925] psia
Primary Coolant Temperature (Design)	[550]°F
(Operating)	[497]°F
Primary Coolant Moisture	1.0 to 10,000 ppmv
Primary Coolant Flow Rate	[1,246,000] lb/h

2. DESIGNER'S ALTERNATIVES

2.1 Use commercial industrial moisture monitoring equipment in the 4 x 350 MW(t) HTGR design without design verification.

3. SELECTED DESIGN APPROACH AND EXPLANATION

It is recommended that sample rake designs, commercial moisture monitors, and auxiliary sample equipment be purchased and/or designed. These candidate moisture monitor systems designs should be compared to confirm accuracy and response times and the selected system verified. This results in a high degree of confidence in the overall selected system and its performance characteristics in contrast to the alternative approach which would not provide for prior design validation.

4. SCHEDULE REQUIREMENTS

Design verification must be completed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Use a moisture monitor design without design verification. This would involve increased risk that the design chosen without verification would not meet plant design requirements relative to safety and/or availability. This would adversely impact both plant schedule and cost.

[signature] 3/12/87
Originator Date

[signature] 031387
Department Manager Date

[signature] for
G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/25/87

VERIFY SHUTDOWN COOLING HEAT EXCHANGER LEAK DETECTION
DDN M.32.02.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 32

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

A shutdown cooling heat exchanger leak can allow water ingress to the primary coolant or helium egress depending upon the mode of plant operation. Appropriate leak detection is necessary for use in the plant protection system. The purpose of this DDN is to compare various leak detection methods to confirm necessary performance characteristics.

1.1 Summary of Function/Title/Assumptions

F2.1.4.1.1.2.2.4.1 "Sense Parameters"/Assumption 1: An adequate sensor methodology can be designed for detecting shutdown heat exchanger leaks.

1.2 Current Data Base Summary

Since Fort St. Vrain does not have a shutdown cooling system, no shutdown cooling heat exchanger leak detection data is available.

1.3 Data Needed

Confirmation that the shutdown cooling heat exchanger (SCHE) leak detection system is adequate to detect moisture ingress events and to detect radioactive helium from escaping to the shutdown cooling water. The response time and accuracy of the leak detector needs to be determined.

Quality Assurance must be in accordance with QAL II.

1.4 Design Parameters/Service Conditions

Shutdown Cooling Heat Exchanger Leak Detector Parameters:

Leak detection accuracy.

Time of response from onset of leak under all extremes of service conditions.

Leak sensor time constant.

Service Conditions

Primary Coolant: Helium

Primary Coolant Pressure (Design): [1041] psia
(Operating): [14 to 925] psia

Primary Coolant Temperature (Operating): [650]°F

Primary Coolant Moisture: 1.0 to 10,000 ppmv

Primary Coolant Flow Rate: [92,000] lb/h

Shutdown Cooling Heat Exchanger Cooling Water Temperature:
(Inlet) [120]°F
(Outlet) [450]°F

Shutdown Cooling Heat Exchanger Cooling Water Pressure: [758] psia

Shutdown Cooling Heat Exchanger Cooling Water Flow:
[232,000] lb/h

2. DESIGNER'S ALTERNATIVES

2.1 Use the SCHE leak detection system without design verification.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The recommended approach consists of design verification of shutdown cooling heat exchanger (SCHE) leak detection systems. The concern is that no method has been previously designed to detect SCHE tube leaks under all SCHE operating conditions.

Since Fort St. Vrain does not have a shutdown cooling system, the SCHE moisture detection/primary coolant ingress detection system has not been used before. Installing a new design without design verification creates the risk of not meeting the design requirements for protection (2.1).

4. SCHEDULE REQUIREMENTS

Design verification testing must be completed by the end of Preliminary Design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to use the SCHE leak detection system without design verification. This may result in the SCHE leak detection system not meeting design requirements. This may delay plant startup and, therefore, adversely impact plant schedule and cost.

Originator 3/13/87 Date

Department Manager 031387 Date

for G. C. Bramblett 3.25.87

Manager, Project Operations Date

DATE: 2/25/87

VERIFY HELIUM MASS FLOW INSTRUMENTATION
DDN M.32.02.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 32

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The plant protection system initiates a reactor trip to prevent fuel damage when a reactor power/primary coolant flow mismatch occurs. The purpose of this DDN is to determine the performance characteristics of the helium mass flow measurement instrumentation.

1.1 Summary of Function/Title/Assumptions

F2.1.1.2.4.1 "Sense Parameters"/Assumption 1: Primary coolant helium mass flow instruments of sufficient accuracy and sensitivity are available.

F2.1.4.1.1.2.1.1.2.1.1 "Sense Parameters"/Assumption 1: Primary coolant helium mass flow instruments of sufficient accuracy and sensitivity are available.

F3.1.1.2.1.1.2.1.1.1.1 "Seismic Parameters"/Assumption 1: Primary coolant helium mass flow instruments of sufficient accuracy and sensitivity are available.

1.2 Current Data Base Summary

The current data base for helium mass flow measurement in HTGRs is based on Fort St. Vrain experience. This measurement at Fort St. Vrain is based on measuring differential pressure at the circulator inlet vanes, is not used in the protection system, and it is not calibrated throughout the primary coolant flow range. At Fort St. Vrain the primary coolant flow rate used for protection system trips is based on the measurement of circulator speed. The German THTR uses a differential pressure measurement at the circulator inlet venturi.

1.3 Data Needed

Verification of Plant Protection and Instrumentation System helium mass flow instrumentation calibration and time constant at plant service conditions.

Quality Assurance must be in accordance with QAL I.

1.4 Data Parameters/Service Conditions

Helium Mass Flow Instrumentation Data Parameters:

Calibration curve from [3.5] lbm/s to [385] lbm/s at plant service conditons.

Helium mass flow instrumentation sensor time constant.

Service Conditions:

Primary Coolant	Helium
Primary Coolant Pressure	
(Design)	[1041] psia
(Operating)	[925] psia
Primary Coolant Temperature	
(Design)	[550]°F
(Operating)	[497]°F
Primary Coolant Moisture	1.0 to 10,000 ppmv
Primary Coolant Flow Rate	[1,246,000] lb/h

2. DESIGNER'S ALTERNATIVES

2.1 Use a helium mass flow measurement system without design verification and rely on calculated flow calibration.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The recommended approach consists of performing design verification of candidate methods for measuring helium mass flow. This results in a high degree of confidence in the accuracy of flow measurement data. The data will be used by the designers to select the best method for direct determination of helium mass flow rate. Additionally the data will provide confirmation of the direct flow measurement approach under small changes in primary helium pressure. The designers will use the data to verify flow element, instrument and transducer sensitivity, resolution and repeatability for use in the Plant Protection and Instrumentation System design and for use in establishing a basis for flow calibration.

The use of a helium mass flow measurement system without design verification would require low tolerance setpoints because a calculated flow calibration would have more uncertainty. This could adversely affect plant availability by causing spurious trips during transients.

4. SCHEDULE REQUIREMENTS

Design verification must be completed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to use a helium mass flow measurement system without design verification and rely on calculated flow calibration.

The consequences are that more conservative assumptions must be utilized in performing plant dynamic analysis and establishing trip setpoints. This has a general deterimental effect on overall plant performance.

AB Zylugzynski 3/12/87
Originator Date

[signature] 031387
Department Manager Date

[signature] for G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/25/87

VERIFY STEAM GENERATOR INLET HELIUM TEMPERATURE INSTRUMENTATION
DDN M.32.02.04
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 32

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The plant protection system initiates a reactor trip on high helium temperature to protect the steam generator as well as act as a backup measurement for high reactor power. The purpose of this DDN is to determine the performance characteristics of a suitable steam generator inlet helium temperature measurement.

1.1 Summary of Function/Title/Assumptions

F2.1.2.5.1 "Sense Parameters"/Assumption 1: Steam generator inlet helium temperature sensors adequate to meet sensor response time and accuracy requirements are feasible.

F2.1.4.1.1.2.2.2.1 "Sense Parameters"/Assumption 1: Steam generator inlet helium temperature sensors adequate to meet sensor response time and accuracy requirements are feasible.

F3.1.1.2.2.1.1 "Sense Parameters"/Assumption 1: Steam generator inlet helium temperature sensors adequate to meet sensor response time and accuracy requirements are feasible.

1.2 Current Data Base Summary

No current data base for measurement of steam generator inlet helium temperature in the steel vessel side-by-side HTGR configuration is available.

1.3 Data Needed

Verification of the Plant Protection and Instrumentation System steam generator inlet helium temperature instrumentation calibration and response time at plant service conditions.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

Steam Generator Inlet Helium Temperature Instrumentation Data Parameters:

Calibration curve from [1000]°F to [1500]°F at plant service conditions.

Instrumentation time constant at plant service conditions.

Service Conditions:

Primary Coolant	Helium
Primary Coolant Pressure	
(Design)	[1041] psia
(Operating)	[925] psia
Primary Coolant Temperature	
(Design)	[1300]°F
(Operating)	[1268]°F
Primary Coolant Moisture	1.0 to 10,000 ppmv
Primary Coolant Flow Rate	[1,246,000] lb/h

2. DESIGNER'S ALTERNATIVES

2.1 Use steam generator inlet helium temperature instrumentation without design verification and rely on analytical calculations to calibrate the temperature sensors.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Design verification of the steam generator inlet helium temperature instrumentation is recommended and consists of evaluation to verify the sensitivity, response time, resolution, and repeatability of the instrumentation and thermowell assembly. Such verification would provide a higher degree of confidence and better basis for the steam generator inlet helium temperature measurement performance characteristics than would alternative 2.1. The data would provide the designers with enough information to verify the design and to calibrate the sensor arrangement.

4. SCHEDULE REQUIREMENTS

Design verification must be completed by the end of preliminary design (9/89).

5. PRIORITY

Urgency: 1
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to use steam generator inlet helium temperature instrumentation without design verification. This solution may result in not meeting the design requirements. This could delay startup and require additional PPIS design efforts.

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Originator Date

Donald A Sleidy 031387
Department Manager Date

[illegible] for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/25/87

VERIFY PLANT PROTECTION AND INSTRUMENTATION SYSTEM (PPIS) SURVEILLANCE TESTING
DDN M.32.02.05
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 32

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The plant protection system requires on-line surveillance testing. The purpose of this DDN is to demonstate that appropriate surveillance testing can be performed that does not adversely impact protection system reliability or plant availability.

1.1 Summary of Function/Title/Assumptions

F.2.1.1.2.4.2 "Command Action"/Assumption 1: PPIS surveillance testing does not cause spurious system trips.

Note: This same assumption appears in numerous other "Command Action" functions.

1.2 Current Data Base Summary

Protection system on-line surveillance testing is required. Current nuclear plants often experience spurious system trips during surveillance testing which results in an adverse effect on plant availability. The primary thrust on existing protection system testing is to uncover "unsafe" failures in the system.

1.3 Data Needed

Demonstrate that PPIS surveillance testing can be performed in an effective manner that will uncover both "safe" and "unsafe" failed components and in so doing not cause spurious system trips that would adversely affect plant availability.

Quality Assurance must be in accordance with QAL I.

1.4 Data Parameters/Service Conditions

Data Required: "Safe" and "Unsafe" failure of any component can be effectively discovered with the proposed surveillance testing features without causing a trip at the system level.

Operating Environment: Temperature: 70°F ± 5°F
Pressure: Atmospheric (Air)
Relative Humidity: 50% ± 5%
Radiation: Background

2. DESIGNER'S ALTERNATIVES

2.1 Accept the PPIS logic and surveillance testing without design verification testing.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The recommended approach is to demonstrate a 2 out of 4 circuit with associated electronic equipment to simulate four redundant channels (one PPIS protection parameter group) can have surveillance testing performed while simulating channel failures, electronic chip failures, etc. and not cause spurious system trips.

This approach provides a firm basis for accepting the PPIS logic and surveillance testing. Accepting the surveillance testing without a design verification program may adversely affect plant availability if a path for introducing spurious trips during performance of surveillance testing has been overlooked. Performing design verification early in the design will allow any necessary corrections to be performed in a cost effective manner.

4. SCHEDULE REQUIREMENTS

The data will be needed near the end of preliminary design and the results documented before the start of final design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to accept the PPIS logic and surveillance testing without the benefit of design verification testing. The consequences of nonexecution are that it would be necessary to modify the PPIS if surveillance testing does not meet reliability and availability requirements for the 4 x 350 MW(t) plant.

This result could cause licensing problems, plant startup delays, adversely impact plant availability and/or require changes in installed equipment.

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Manager, Project Operations Date

DATE: 2/27/87

FUEL HANDLING MACHINE (FHM), HANDLING MECHANISM AND GRAPPLE DESIGN VERIFICATION
DDN M.34.13.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 34

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel handling machine (and its components) must be highly reliable in order to perform the refueling sequences in the scheduled time.

1.1 Summary of Function/Title/Assumptions

F.1.3.5, "Refuel Core," Assumption 1: Operability and reliability of the refueling mechanisms are sufficient to meet availability requirements for the plant. Individual mechanisms (fuel transfer cask, fuel handling machine, and plug actuator) meet individual reliability goals, and overall system meets its reliability goal.

1.2 Current Data Base Summary

The FHM conceptual design is based on the refueling equipment at Fort St. Vrain and the various large HTGR designs developed over the past 20 years. The FHM mechanisms differ from the Fort St. Vrain assembly as follows:

- o Shorter grapple probe.
- o Electrically controlled grapple mechanisms rather than pneumatic.
- o Handling mechanism linkage radial displacement increased.
- o Viewing system and electronic control system revised to incorporate current technology.
- o Grapple redesigned to interface with prismatic elements.

Years of experience with the FSV FHM have demonstrated reliable features of the design and some which could be improved. The differences listed here are proposed as improvements, but must be tested to verify that they are as good as or better than the FSV FHM.

1.3 Data Needed

Data are needed on functional and performance limits in all anticipated operating modes in order to establish the operability and reliability of all components under expected environmental conditions. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal atmosphere	Helium/air
Pressure	14.7 psia
Temperature	[TBD]
Helium inlet gas temperature	240°F (shutdown)
Hoist speed range	2 to 24 in./s

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Test the equipment at the site during preoperational checkout.

2.2 Test the equipment during system integration test.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A full scale test rig and test article will be used to obtain early reliability (life) data for the machine and its subcomponents. This will provide an opportunity to correct deficiencies so that the assembly will function satisfactorily in all operating modes during the system qualification test. Alternatives 2.1 and 2.2 carry substantial risk of schedule delay because of the discovery of problems late in the schedule. The selected approach reduces the potential for schedule delay because the problems are identified earlier and, therefore, can be fixed earlier.

4. SCHEDULE REQUIRMENTS

Testing to be completed 12 months prior to completion of the plant final design phase (9/92).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is Alternative 2.2. Nonexecution of the preferred approach would lead to total dependence on Alternative 2.2, and failure at that time would most certainly lead to schedule delays while the problems were investigated and corrected.

Edwin C. Haney 3/25/87

Originator Date

R F Turner 3/25/87

Department Manager Date

for G. C. Bramblette 3.25.87

Manager, Project Operations Date

DATE: 2/27/87

FUEL TRANSFER CASK COMPONENT DESIGN VERIFICATION
DDN M.34.13.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 34

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel transfer cask (and its components) must be highly reliable in order to perform the refueling sequences in the scheduled time.

1.1 Summary of Function/Title/Assumptions

F.1.3.5, "Refuel Core," Assumption 1: Operability and reliability of the refueling mechanism are sufficient to meet availability requirements for the plant. Individual mechanisms (fuel transfer cask, fuel handling machine, and plug actuator) meet individual reliability goals, and overall system meets its reliability goal.

1.2 Current Data Base Summary

The fuel transfer cask is an entirely new machine for which there are no data even though the design is similar to the cask design developed for earlier HTGRs.

1.3 Data Needed

There are several mechanisms within the assembly which must be evaluated. These include the vertical drive system for the hoist grapple, horizontal transfer table drive, and the complete grapple system.

Data are needed on functional and performance limits in all anticipated operating modes in order to establish the operability and reliability of all components under expected environmental conditions.

Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal atmosphere	Helium/air
Pressure	14.7 psia
Temperature	[TBD]
Helium inlet gas temperature	240°F (shutdown)
Hoist speed range	2 to 24 in./s

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Test the equipment at the site during preoperational checkout.

2.2 Test the equipment during system integration test.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A full-scale test rig and test article will be used to obtain early reliability (life) data for the machine and its subcomponents. This will provide an opportunity to correct deficiencies so that the assembly will function satisfactorily in all operating modes during the system qualification test. Alternatives 2.1 and 2.2 carry substantial risk of schedule delay because of the discovery of problems late in the schedule. The selected approach reduces the potential for schedule delay because the problems are identified earlier and, therefore, can be fixed earlier.

4. SCHEDULE REQUIRMENTS

Testing to be completed 12 months prior to completion of the plant final design phase (9/92).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The preferred fallback position is Alternative 2.2. Nonexecution of the preferred approach would lead to total dependence on Alternative 2.2 and failure at that time would most certainly lead to schedule delays while the problems were investigated and corrected.

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Originator Date

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PLUG ACTUATOR AND TURNTABLE ASSEMBLY COMPONENT DESIGN VERIFICATION
DDN M.34.13.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 34

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The plug actuator and turntable assembly (PA&T) must be highly reliable in order to perform the refueling sequences in the scheduled time.

1.1 Summary of Function/Title/Assumptions

F.1.3.5, "Refuel Core," Assumption 1: Operability and reliability of the refueling mechanism are sufficient to meet availability requirements for the plant. Individual mechanisms (fuel transfer cask, fuel handling machine, and plug actuator) meet individual reliability goals, and overall system meets its reliability goal.

1.2 Current Data Base Summary

The plug actuator and turntable assembly is an entirely new machine for which there are no data even though the plug removal design is similar to equipment developed for earlier HTGRs.

1.3 Data Needed

There are several mechanisms within the assembly which must be evaluated. These include the plug removal mechanism, the gate drive, and the turntable drive.

Data are needed on functional and performance limits in all anticipated operating modes in order to establish the operability and reliability of all the components. Quality assurance must be in accordance with the requirements for Quality Assurance Level II.

1.4 Data Parameters/Service Conditions

Internal atmosphere	Helium/air
Pressure	14.7 psia
Temperature	[TBD]

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Test the equipment at the site during preoperational checkout.

2.2 Test the equipment during system integration test.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A full-scale test rig and test article will be used to obtain early reliability (life) data for the machine and its subcomponents. This will provide an opportunity to correct deficiencies so that the assembly will function satisfactorily in all operating modes during the system qualification test. Alternatives 2.1 and 2.2 carry substantial risk of schedule delay because of the discovery of problems late in the schedule. The selected approach reduces the potential for schedule delay because the problems are identified earlier and, therefore, can be fixed earlier.

4. SCHEDULE REQUIRMENTS

Testing to be completed 12 months prior to completion of the plant final design phase (9/92).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The preferred fallback position is Alternative 2.2. Nonexecution of the preferred approach would lead to total dependence on Alternative 2.2, and failure at that time would most certainly lead to schedule delays while problems were investigated and corrected.

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VERIFY FUEL HANDLING SYSTEM INSTRUMENTATION & CONTROLS
DDN M.34.13.05
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 34

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The Fuel Handling Control System requires rapid and positive identification of fuel elements during remote fuel handling. The purpose of this DDN is to determine performance characteristics of the selected system and ensure control system compatibility with the fuel handling mechanisms.

1.1 Summary of Function/Title/Assumptions

F1.3.5.4 "Account for Elements"/Assumption 1: Elements are identifiable remotely.

F1.3.5 "Refuel Core"/Assumption 1: Operability and reliability of refueling mechanisms are sufficient to meet the availability requirement for the plant. Individual mechanisms (FTC, FHM and Plug Actuator) meet individual reliability goals, and overall system meets its reliability goal.

F1.3.5 "Refuel Core"/Assumption 5: Elements are identifiable remotely.

1.2 Current Data Base Summary

The current data base is Fort St. Vrain experience, large HTGR designs and industrial applications for computer controlled equipment.

1.3 Data Needed

1. Bounding values of factors (element motion, direction, velocity, size of identification marking, temperature, etc.) which cause failures in serial number identification).

2. Recovery time (time to repair plus time to get back to automatic operation) for each failure mode.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

Console and Electronics Cabinets

Atmosphere: Air
Temperature: [72]°F
Pressure: Atmospheric
Relative Humidity: [10-90%]

Components in Reactor Vessel

Temperature: [240]°F
Pressure: 14.7 psia
Atmosphere: Helium/Air

2. DESIGNER'S ALTERNATIVE

2.1 Verify performance of computer control software and instruments only during assembly/checkout of Fuel Handling Control Station.

2.2 Verify performance of subsystems of computer control and instruments during development and systems integration tests (DDN M.34.13.01 through .04).

3. SELECTED DESIGN APPROACH AND EXPLANATION

It is recommended that the performance and environmental compatibility of components and subsystems be verified to firm up design prior to the overall system development and reliability verification. Early confirmation of performance and compatibility of control software and instruments is needed to support the design of machines and of the control system to reduce potential delays in performing the mechanical equipment development and system integration tests (2.2) (DDN M.34.13.01 through .04).

4. SCHEDULE REQUIREMENTS

Verification data should be available by the end of the preliminary design phase (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to verify performance of the subsystems of computer control and instruments during the development and system integration tests. If prior verification is not performed, some schedule delay and probable control system redesign during performance, reliability and integrated system tests may be experienced (DDN M.34.13.01 through .04).

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Originator Date

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Department Manager Date

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Manager, Project Operations Date

DATE: 2/25/87

VERIFY NSSS CONTROL SYSTEM WITH SIMULATOR
DDN M.37.01.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 37

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The NSSS control for the plant will incorporate microprocessor based controllers and the control schemes programmed into these controllers need to be validated. The NSSS control schemes require revision and the NSSS control modules require retuning as the NSSS control system design evolves. The purpose of this DDN is to verify proper dynamic operation of NSSS controls from the main control room, including checkout of NSSS control modules and control schemes. Additionally, the interaction of the control room operator with NSSS controls will be verified to assure controllability design goals are met.

1.1 Summary of Function/Title/Assumptions

F1.1.8.3.1 "Make Energy Production Decisions"/Assumption 1: NSSS controls can be designed and built so the NSSS can be automatically and/or manually controlled over the operating range and the operator can interact with the NSSS to make operating decisions.

F1.2.8.3.1 "Make Energy Production Decisions"/Assumption 1: NSSS controls can be designed and built so the NSSS can be automatically and/or manually controlled over the operating range and the operator can interact with the NSSS to make operating decisions.

F1.3.8.3.1 "Make Energy Production Decisions"/Assumption 1: NSSS controls can be designed and built so the NSSS can be automatically and/or manually controlled over the operating range and the operator can interact with the NSSS to make operating decisions.

F1.4.8.3.1 "Make Energy Production Decisions"/Assumption 1: NSSS controls can be designed and built so the NSSS can be automatically and/or manually controlled over the operating range and the operator can interact with the NSSS to make operating decisions.

F2.1.8.3.1 "Make Energy Production Decisions"/Assumption 1: NSSS controls can be designed and built so the NSSS can be automatically and/or manually controlled over the operating range and the operator can interact with the NSSS to make operating decisions.

1.2 Current Data Base Summary

NSSS computer analysis of representative transients.

Computer analysis currently does not reflect characteristics of yet to be selected hardware and software.

1.3 Data Needed

Verification that the NSSS control digital hardware and software is capable of controlling the NSSS throughout the plant operating range.

Verification that the NSSS control man-machine interface meets human factors criteria under all plant operating conditions.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

NSSS Controls Data Parameters:

NSSS controls hardware and software performance parameters: time of response, overshoot, settling time, stability, and steady-state error.

NSSS controls man-machine interface performance parameters: operator time of response, NSSS control system time of response to all types of operator inputs, control system operator feedback delay times, operator error rate, manual control overshoot, settling time, stability, and steady-state error.

Service Conditions:

Temperature:	[75]°F
Relative Humidity:	[50]%
Pressure:	Atmospheric

2. DESIGNER ALTERNATIVE

2.1 Use the actual plant during startup testing to verify the NSSS analog and microprocessor based control.

3. SELECTED DESIGN APPROACH AND EXPLANATION

It is recommended that the plant NSSS analog and digital control and NSSS control schemes be verified by using a plant simulator. FSV experience has shown that testing control schemes with the plant during startup testing causes numerous equipment trips. These equipment trips use a portion of the design life of some components (e.g., circulators, steam generators, etc.). Verifying control schemes prior to plant startup testing can reduce the number of equipment transients and trips encountered during plant startup testing.

4. SCHEDULE REQUIREMENTS

Design verification must be completed before the end of construction. (9/95).

5. PRIORITY

Urgency: 4
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is that all NSSS control testing would have to be performed during plant startup testing. This will probably cause additional equipment transients and trips and may increase the time required to perform plant startup testing. Some control redesign might be required to meet NSSS controllability goals.

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G. C. Bramblett 3.25.87
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DATE: 2/25/87

VERIFY NSSS CONTROL LAYOUT
DDN M.37.01.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 37

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The plant operator/NSSS control system interface is important in assuring that the plant operator can control the NSSS during normal and abnormal conditions. This DDN is to establish the adequacy of the main control room NSSS instrumentation and control layouts.

1.1 Summary of Function/Title/Assumptions

F1.1.8.1.1 "Accept Energy Production Direction"/Assumption 1: The NSSS Control Man-Machine Interface (MMI) can be designed and built to accept energy production direction.

1.2 Current Data Base Summary

Industry human factors criteria data and human factors evaluation techniques.

1.3 Data Needed

Adequacy of main control room and remote shutdown area NSSS instrumentation and control layouts. Operator times to accomplish NSSS energy production direction.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

NSSS Control Man-Machine Interface Data Parameters:

Task analysis worksheets for NSSS control function throughout the operating range.

Service Conditions:

Physical Space:	[40 x 60 feet]
Temperature:	[75]°F
Relative Humidity:	[50]%
Pressure:	Atmospheric

2. DESIGNER'S ALTERNATIVES

2.1 Perform the NSSS control design without the benefit of mock-ups.

3. SELECTED DESIGN APPROACH AND EXPLANATION

It is recommended that a mockup of the main control room and remote shutdown room layout of NSSS equipment be established and that walk-through simulations of typical normal and abnormal operating procedures be performed. This would verify that appropriate human engineering criteria are met. Performing the design without the benefit of markups may require redesign at a later date if human engineering criteria are not met.

4. SCHEDULE REQUIREMENTS

Design verification of NSSS controls layout should be completed prior to starting construction of a main control room simulator facility (9/90).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to perform the NSSS control panel layout without using a mock-up. This may require redesign at a later date if human engineering criteria are not met.

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Manager, Project Operations Date

DATE: 2/25/87

VERIFICATION OF CRT DISPLAYS OF NSSS MODULE OPERATING DATA
DDN M.37.01.03
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 37

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The plant operator/NSSS operating data interface is important in assuring the reactor operator can readily assimilate the module status. This DDN is to determine which displays are most effective in conveying data to the operator.

1.1 Summary of Function/Title/Assumptions

F1.1.8.5.1 "Report Energy Production Information"/Assumption 1: Displays of NSSS production information provide operating parameters meaningful to the plant operations staff.

1.2 Current Data Base Summary

General industry human factors criteria.

1.3 Data Needed

Verification that NSSS module CRT displays meet human factors criteria.

Quality Assurance must be in accordance with QAL II.

1.4 Data Parameters/Service Conditions

NSSS Module CRT Display Data Parameters:

NSSS module CRT parameters: color, intensity, luminance contrast, character size, regeneration rate, viewing distance, screen luminance, luminance range, ambient luminance, reflected glare, readability, geometric distortion, resolution, and color layout.

Service Conditions:

Temperature:	[75]°F
Relative Humidity:	[50]%
Pressure:	Atmospheric

2. DESIGNER'S ALTERNATIVES

2.1 Create displays of NSSS module parameters and assume their adequacy without verification.

3. SELECTED DESIGN APPROACH AND EXPLANATION

It is recommended that dynamic displays of NSSS module operating parameters be created and that walk throughs of typical startup, shutdown, normal operation, and accident procedures, with dynamic updating of the displays. This would verify the human engineering of these displays and ensure providing the maximum assimilation of information to the operator.

4. SCHEDULE REQUIREMENTS

Design verification completed before the end of preliminary design (9/89).

5. PRIORITY

Urgency: 2
Cost benefit: L
Uncertainty in existing data: L
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position is to create displays of NSSS module parameters and assume their adequacy until plant startup. If the CRT displays of NSSS conditions are verified for adequacy later in the overall program, this could cause schedule slippage. Omission of display verification altogether could result in increased risk of plant operator error.

AB Zgliczynski 3/13/87
Originator Date

John A Blady 031387
Department Manager Date

Oullimoon for
G. C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

SHUTDOWN CIRCULATOR MOTOR COOLING DESIGN VERIFICATION
DDN M.57.01.01
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 57

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Data is required to verify the thermal/hydraulic performance of the shutdown circulator motor cooling system.

1.1 Summary of Function Number/Title/Assumptions

F2.1.2.2.4.3.2.1.2.2. "Power Circulator" Assumption 1: Submerged motor cooling will be verified.

1.2 Current Data Base Summary

No experimental data is available for submerged motor cooling in helium.

1.3 Data Needed

Applicable cooling flow rate, temperature, pressure and rotational speed measurements are needed to ascertain thermal/hydraulic performance of the motor cooling system, including motor cooling passages, motor cooling fans and motor water/helium heat exchanger. Cooling of the enclosed motor is essential. Because of the potential impact on the overall circulator configuration and performance, it is important to confirm the capability of the motor cooling system with this separate test early in the design phase.

Quality assurance must satisfy QAL II requirements.

1.4 Data Parameters/Service Conditions

Experimental data will be obtained with the full scale motor. Heat generated by the operating motor will be removed by forced convection, provided by fans, along the motor shaft. Water cooling tubes around the motor will dissipate the heat to an external sink. The following ranges of test conditions will provide data sufficient to extrapolate data to design conditions.

Pressure (psia)	Ambient - [18]
Temperature (°F)	Ambient - [300]
Water Pressure (psia)	[50-100]
Water Temperature (°F)	[50-80]
Water Flow (gpm)	[10-25]
Drive Power (hp)	[15]
Heat Load (kW)	[10]

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Rely on analysis and confirm analytical predictions during shutdown circulator prototype test (M.57.01.02).

2.2 Rely on data from main circulator test (M.21.01.03).

3. SELECTED DESIGN APPROACH AND EXPLANATION

A test simulating flow conditions was selected to obtain the required data, because the alternative of testing during the prototype test could cause severe schedule risk if equipment does not perform as predicted. It may be possible to combine this test with the main circulator test (Alternative 2.2). This can only be determined after more detailed designs have been accomplished.

4. SCHEDULE REQUIREMENTS

Data from this test is required six months prior to start of manufacture of the circulator for the Prototype Shutdown Circulator Test (9/91).

5. PRIORITY

Urgency: 3
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Data could be obtained from the prototype test (2.1) approximately 2 years later. Failure to confirm the design at that time could result in schedule delays.

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Originator Date

R F Turner 3/12/87
Department Manager Date

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G.C. Bramblett 3.25.87
Manager, Project Operations Date

DATE: 2/27/87

SHUTDOWN CIRCULATOR PROTOTYPE DESIGN VERIFICATION
DDN M.57.01.02
PROJECT NUMBER 6300

PLANT: 4 x 350 MW(t) Modular HTGR/System 57

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Data is required to verify the capability of the entire shutdown circulator subsystem to provide adequate primary coolant circulation for various plant operating requirements. Because the circulator operating time and number of cycles represent a relatively small percentage of plant life, data will also be obtained to predict the effect of the circulator and its associated systems on plant availability.

1.1 Summary of Function Number/Title/Assumptions

F2.1.2.2.4 "Establish Conditions for Circulator Repair" Design Selection 2: Probability of shutdown cooling loop operating $\geq$[99]%.

F2.1.2.2.4.3.2.1.2 "Pump Primary Coolant" Assumption 1: Compressor/shutoff valve performance and interaction have been verified.

1.2 Current Data Base Summary

The data base applicability is limited to the design of individual components such as the centrifugal compressor and magnetic bearings. It is inadequate to verify performance and reliability of the shutdown circulator because no shutdown circulator cooling system testing has been done, and the combined configuration of electric motor drive, centrifugal flow compressor, loop shutoff valve, inlet and diffuser is unique to the current subsystem. No GA data base exists for verification of centrifugal compressor performance.

1.3 Data Needed

Applicable flow rate, temperature, pressure, speed, vibration and sound measurements are required to verify the performance of the circulator and its associated systems for all anticipated reactor operating conditions. Testing to include verification of interaction of the shutdown cooling control system with the circulator systems.

Quality assurance must satisfy QAL II requirements.

1.4 Data Parameters/Service Conditions

Verification to be performed in helium with full scale prototype hardware including circulator, ducting, loop shutoff valve, service system, instrumentation, motor and control. The following ranges of test conditions will envelope depressurized and pressurized design conditions:

Exit Pressure, (psia)	Ambient - 925
Inlet Temperature, (°F)	Ambient - 600°
Helium Pressure Rise, (psi)	0 - 0.71
Helium Flow, (lb/h)	0 - 22,900
Shaft Power, (hp)	0 - [215]
Reference Transients	Pressurized startup and shutdown of circulator Depressurized startup and shutdown of circulator Depressurization with motor stopped

2. DESIGNER'S ALTERNATIVES

The following alternatives are available:

2.1 Verify equipment performance after installation in the reactor vessel.

2.2 Perform subassembly tests only.

3. SELECTED DESIGN APPROACH AND EXPLANATION

A complete prototype subsystem test was selected to obtain the required data because the alternative of testing after installation of equipment in the reactor vessel (2.1) could extend plant acceptance schedule and cause severe schedule and cost risk if equipment does not perform as predicted. Subassembly tests only (2.2) would not provide adequate assurance against potential schedule and cost risks.

4. SCHEDULE REQUIREMENTS

Completion of prototype test is required prior to release of hardware production drawings (9/93).

5. PRIORITY

Urgency: 3
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

Alternative (2.2) above is the initial fallback position. Failure to confirm the design could result in schedule and cost delays if the production hardware does not perform acceptably during hot flow tests.

M. K. Nichols 2-27-87

Originator Date

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Department Manager Date

[illegible] for

G.C. Bramblett 2.25.87

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