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SNPC-3

Records

MASTER NUCLEAR DIVISION

RN-PA-0028

VOLUME II

WESTINGHOUSE ASTRONUCLEAR LABORATORY

MATERIAL PRESENTED TO SNPO

IN

SHIELDING TRADE STUDY REVIEW (U)

12 December 1969

NERVA Program

Contract SNP-1

Prepared by:

M. A. Capo (WANL)

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VOLUME II

MASTER

SPECIAL REVIEW FINAL DETERMINATION	Review	Class	Date
	AW	U	4-20-82

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MATERIAL PRESENTED TO SNPO
IN
SHIELDING TRADE STUDY REVIEW (U)

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NUCLEAR ROCKET OPERATIONS

Classification cancelled (or changed to)

Prepared by:

M. A. Capo (WANL)

by authority of

by H.F.C. TIC, date SEP 12 1973

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TASK 23.W - WANL STUDY OF INTERNAL SHIELD MATERIAL

1. INTRODUCTION
2. DESCRIPTION OF REFERENCE DESIGN SHIELD
3. COMPARISON OF REFERENCE DESIGN SHIELD RADIATION
LEVELS WITH CURRENT ENGINE SPECIFICATION
4. PARAMETRIC ANALYSIS OF HIGH DENSITY SHIELD MATERIALS
5. CONCLUSIONS AND RECOMMENDATIONS

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DETAILED WEIGHT OF THE REFERENCE DESIGN SHIELD (939J685E)

(THIS PAGE IS UNCLASSIFIED)

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COMPARISON OF CALCULATED RADIATION LEVELS WITH SPECIFICATIONS *
FOR CRITICAL ENGINE COMPONENTS OF R-1 REF. DESIGN

LOCATION	<u>GAMMA KERMA RATE</u> <u>(RAD-C/HR)</u>			<u>FAST NEUTRON FLUX</u> <u>(E>1 MEV) (N/CM²-SEC)</u>		
	SPEC.	CALC.	<u>SPEC.</u> <u>CALC.</u>	SPEC.	CALC.	<u>SPEC.</u> <u>CALC.</u>
<u>LTS - 1</u>	2.8 (7)	4.4 (6)	6.4	1.0 (13)	7.0 (10)	140.0
<u>LTS - 2</u>	2.3 (7)	5.9 (6)	3.9	8.0 (12)	1.0 (11)	80.0
<u>LTS - 3</u>	2.0 (7)	6.5 (6)	3.1	7.0 (12)	1.1 (11)	64.0
<u>LTS - 4</u>	1.6 (7)	7.7 (6)	2.1	6.0 (12)	1.5 (11)	40.0
<u>TPA - 1</u>	4.8 (6)	9.9 (5)	4.8	3.5 (11)	1.9 (10)	18.0
<u>TPA - 2</u>	4.4 (6)	1.2 (6)	3.7	2.6 (11)	2.2 (10)	12.0

* PER AGC TRANSMITTAL OF AUGUST 25, 1969

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COMPARISON OF CALCULATED RADIATION LEVELS WITH SPECIFICATIONS*
FOR CRITICAL ENGINE COMPONENTS OF R-1 REF. DESIGN

LOCATION	<u>GAMMA KERMA RATE</u> <u>(RAD-C/HR)</u>			<u>FAST NEUTRON FLUX</u> <u>(E>1.0 MEV) (N/CM²-SEC)</u>		
	SPEC.	CALC.	<u>SPEC.</u> <u>CALC.</u>	SPEC.	CALC.	<u>SPEC.</u> <u>CALC.</u>
<u>EES - 1</u>	6.3 (6)	2.8 (6)	2.3	4.1 (11)	4.9 (10)	8.4
<u>EES - 2</u>	4.7 (6)	2.5 (6)	1.9	3.2 (11)	4.2 (10)	7.6
<u>EES - 3</u>	4.8 (6)	1.8 (6)	2.7	4.0 (11)	2.9 (10)	14.0
<u>EES - 4</u>	9.2 (6)	4.0 (6)	2.3	4.5 (11)	4.1 (10)	11.0
<u>EES - 5</u>	1.3 (7)	1.5 (7)	0.9	5.1 (11)	4.7 (11)	1.1
<u>TPCV</u>	---	---	---	5.8 (11)	1.4 (10)	41.0

* PER AGC TRANSMITTAL OF AUGUST 25, 1969

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COMPARISON OF CALCULATED RADIATION LEVELS WITH SPECIFICATIONS *
FOR CRITICAL ENGINE COMPONENTS OF R-1 REF. DESIGN

LOCATION	<u>INTERMEDIATE NEUTRON FLUX</u> <u>($0.4\text{EV} < E < 1.0 \text{ MEV}$)</u> <u>(N/CM²-SEC)</u>			<u>THERMAL NEUTRON FLUX</u> <u>($E < 0.4 \text{ EV}$)</u> <u>(N/CM²-SEC)</u>		
	SPEC.	CALC.	<u>SPEC.</u> <u>CALC.</u>	SPEC.	CALC.	<u>SPEC.</u> <u>CALC.</u>
<u>EES - 1</u>	6.6 (11)	9.5 (10)	6.9	4.9 (10)	6.6 (9)	7.4
<u>EES - 2</u>	5.8 (11)	6.1 (10)	9.5	3.8 (10)	4.2 (9)	9.0
<u>EES - 3</u>	7.2 (11)	3.5 (10)	21.0	4.8 (10)	1.9 (9)	25.0
<u>EES - 4</u>	8.1 (11)	2.4 (11)	3.4	5.4 (10)	2.7 (9)	20.0
<u>EES - 5</u>	9.2 (11)	3.0 (12)	0.3	6.1 (10)	3.4 (10)	1.8
<u>TPCV</u>	---	---	---	7.0 (10)	9.2 (8)	76.0

* PER AGC TRANSMITTAL OF AUGUST 25, 1969

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LOCATIONS WHERE R-1 REFERENCE DESIGN (939J685E) CALCULATION EXCEEDS SPECIFICATION *

<u>LOCATION</u>	<u>TYPE OF RADIATION</u>	<u>RATIO OF SPEC/CALC</u>
EES - 5	GAMMA RAY	0.9
	INTERMEDIATE NEUTRON	0.3
TIL - 1	THERMAL NEUTRON	0.9
	INTERMEDIATE NEUTRON	0.5
PDL - 1	THERMAL NEUTRON	0.9
	INTERMEDIATE NEUTRON	0.5
PDL - 2	INTERMEDIATE NEUTRON	0.9
TANGENT PLANE (R = 0" TO R = 15") GAMMA RAY		0.7
TANGENT PLANE (R = 40" TO R = 50") GAMMA RAY		0.3
TANGENT PLANE (R = 50") INTERMEDIATE NEUTRON		0.8

* PER AGC TRANSMITTAL OF AUGUST 25, 1969

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HIGH DENSITY PARAMETRIC SHIELD STUDY MATERIALS INVESTIGATED

- BORATED TITANIUM HYDRIDE
- BORATED STEEL + BORATED TITANIUM HYDRIDE
- LEAD + BORATED TITANIUM HYDRIDE
- LEAD
- BORATED STEEL

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SUMMARY OF ANALYSIS

ΔWEIGHT AND ΔTHICKNESS COMPARED TO THE REFERENCE DESIGN

SHIELD COMPOSITION (% VOLUME)	CONDITION A		CONDITION B	
	WEIGHT (LBS.)	THICKNESS (IN.)	WEIGHT (LBS.)	THICKNESS (IN.)
(80% BTiH _{1.5} , 20% BSS)	+ 30	-5.88	+ 110	-5.58
(50% BTiH _{1.5} , 50% BSS)	+ 875	-6.02	+ 915	-5.9
(20% BTiH _{1.5} , 80% BSS)	+ 2080	-5.15	+2200	-4.95
(100% BTiH _{1.5})	- 160	-4.05	- 70	-3.7
(100% BSS)	+ 3870	-2.75	+4020	-2.45

NOTE: Condition A - Detector location fixed with respect to core center (Preliminary Engine Spec + 15 inches in axial direction)

Condition B - Detector location fixed with respect to the variable forward surface of shield

Reference Design (WANL Dwg. 939J685E)	} Central Shield Assembly Thickness - 18.25" Weight (per this study) - 3430 pounds
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W58835-1

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W59840

FAST NEUTRON FLUX ($E > 1.0$ Mev) AT EES-5 (R=45 INCHES) VS. SHIELD MASS THICKNESS

10¹³

NOTE: For All Configurations EES-5 is Located at a Constant Distance From the Core Midplane (i.e., 253.6 cm). To Convert Abscissa to Shield Weight (lbs.) Multiply by 29.2.

10¹²FAST NEUTRON FLUX ($E > 1.0$ Mev)10¹¹10¹⁰10⁹○ ALL BTiH_{1.5}

□ ALL Pb

▽ 12" BTiH_{1.5}⬡ 9" BTiH_{1.5}◇ 6" BTiH_{1.5}

* REF. DESIGN (DRWG. 939J685 E)

} WITH VARIABLE Pb

SHIELD MASS THICKNESS (GM/CM²)

W58934-1

W59934-1

W59834-1

W58934-1

COMPARISON OF RADIATION LEVELS AT EES-1 AND EES-5 FOR THE
CURRENT ENGINE SPECIFICATION WITH (6" BTiH_{1.5} + 3" Pb) AND
THE REFERENCE DESIGN SHIELDS

DATA POINT	RADIATION TYPE	UNITS	ENGINE SPEC	REFERENCE DESIGN	6" BTiH _{1.5} ^{**} 3" Pb
EES-1 *	GAMMA DOSE RATE	REM/HR	6.3 (6)	2.8 (6)	1.5 (6)
EES-1	NEUTRON FLUX E > 1.0 Mev	N/CM ² -SEC	4.1 (11)	4.9 (10)	3.3 (11)
EES-1	NEUTRON FLUX 0.4ev < E < 1.0Mev	N/CM ² -SEC	6.6 (11)	9.5 (10)	1.5 (12)
EES-1	NEUTRON FLUX E < 0.4 ev	N/CM ² -SEC	4.9 (10)	6.6 (9)	9.0 (10)
EES-5 *	GAMMA DOSE RATE	REM/HR	1.3 (7)	1.5 (7)	1.4 (7)
EES-5	NEUTRON FLUX E > 1.0 Mev	N/CM ² -SEC	5.1 (11)	4.7 (11)	3.9 (11)

NOTE: NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

- * EES-1 IS LOCATED ON THE CENTERLINE OF THE DISC SHIELD.
EES-5 IS LOCATED AT THE OUTER RADIUS OF THE DISC SHIELD.

- ** A NUCLEAR EQUIVALENT SHIELD CONSISTS OF ~ 10 INCHES OF BATH PLUS 3 INCHES OF LEAD.

W50035-1

COMPARISON OF MATERIAL PROPERTIES

BATH VS BORATED TiH

PROPERTY * (ROOM TEMPERATURE)	BATH	TiH
T _s ULT psi	18,000	15,000
T _s Y psi	12,000	5000
Elongation %	2.75%	1/2 - 1%
C _s Y psi	13,500	-
E psi	9.555×10^6	13.0×10^6
ρ Lb/In ³	.103	.137
Coef. Lin. Exp In./In. - °R	7.75×10^{-6}	6.0×10^{-6}
Coef. Th. Cond. BTU/FT/Hr-FT ² -°F	74.0	11.0
Specific Heat BTU/LB	.20	-
Melting Pt. °R	1691	-

* Note: THESE PROPERTIES ARE AVAILABLE AS $\bar{X}$ VALUES WITH STANDARD DEVIATION FOR BATH IN A RANGE OF TEMPERATURE 140 - 1080°R. (TME-1947, AUGUST 69)

THE PUBLISHED DATA ON TiH IS VERY LIMITED AND IS GENERALLY SHOWN AS 20° - 320°C. FOR THIS REASON ONLY ROOM TEMPERATURE DATA IS TABULATED FOR COMPARISON.

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STRUCTURAL EVALUATION OF LEAD, TiH FORWARD SHIELD

ESTIMATED MATERIAL PROPERTIES

	<u>LEAD (322°R)</u>	<u>TiH₂ (512°R)</u>
MODULUS OF ELASTICITY (E)	3.4×10^6 PSI	20×10^6 PSI (EST. MAX.)
COEF. OF THERMAL EXPANSION (α)	$15 \times 10^{-6} \text{ } ^\circ\text{R}^{-1}$	$5 \times 10^{-6} \text{ } ^\circ\text{R}^{-1}$ (EST. MAX)
POISSON'S RATIO (μ)	0.45	0.3
ULTIMATE TENSILE STRENGTH (σ_{ult})	2900 PSI	10,000 PSI (EST. MIN.)
ULTIMATE ELONGATION (ϵ_{ult})	~ 50%	< 1%
TYPE OF MATERIAL	DUCTILE	BRITTLE
TEMPERATURE DIFFERENTIAL = ($T_{AVE} - T_{MIN}$)	6°R	56°R
THERMAL STRAIN = $\alpha (T_{AVE} - T_{MIN})$	0.01%	0.03%
THERMAL STRESS = $E \alpha (T_{AVE} - T_{MIN}) / (1 - \mu)$	560 PSI	8000 PSI

CONCLUSIONS

1. THERE IS SUFFICIENT STRENGTH AND DUCTILITY IN THE LEAD SHIELD FOR THIS APPLICATION. NO PROBLEMS WITH THERMAL STRAINS ARE EXPECTED.
2. BASED ON PRELIMINARY ESTIMATED MAXIMUM $E\alpha$ AND MINIMUM STRENGTH, THE BRITTLE TiH SHIELD WILL NOT CRACK IF THE TEMPERATURE DIFFERENTIALS ARE 56°R OR LESS. BETTER MATERIALS DATA MAY INDICATE A LARGER TOLERABLE DIFFERENTIAL.

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THERMAL EVALUATION
(FULL POWER STEADY STATE)

ASSUMED OR ESTIMATED ENVIRONMENT

	<u>LEAD</u>	<u>BORATED TITANIUM HYDRIDE</u>
MAXIMUM HEAT GENERATION RATE (BTU/HR IN ³)	440	4000
THERMAL CONDUCTIVITY (BTU/HR FT °R)	19.6	11
SHIELD THICKNESS (INCHES)	3.0	6.0
COOLANT FLOW GEOMETRY:		
CHANNEL DIAMETER (INCHES)	0.25	0.219
PITCH (INCHES)	0.60	0.50
TYPE OF ARRAY	TRIANGULAR	SQUARE*
VOID FRACTION	0.16	0.20
BULK COOLANT TEMPERATURE (°R)	300	300
FILM COEFFICIENT (BTU/HR IN ² °R)	14.1	11.2

RESULTS OF CONCEPTUAL ANALYSIS

MAXIMUM MATERIAL TEMPERATURE (°R)	322°R	512°R
TEMPERATURE DIFFERENCE - MAXIMUM TO MINIMUM (°R)	12	112
TEMPERATURE DIFFERENCE - AVERAGE TO MINIMUM (°R)	6	56
FILM TEMPERATURE DIFFERENCE (°R)	10	100
PRESSURE DROP (PSI)	2.3	1.7

* TiH SHIELD IS COMBINATION OF SQUARE AND TRIANGULAR ARRAYS. WITH SQUARE ARRAY GIVING MAXIMUM TEMPERATURES.

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NUCLEAR DESIGN PERFORMANCE CONSIDERATIONS

PERFORMANCE CHARACTERISTIC

- REACTIVITY OF DRY REACTOR AT ROOM TEMPERATURE
- OPERATING REACTIVITY CHANGES DUE TO POWER, CORROSION, BURNUP
- CONTROL DRUM REACTIVITY WORTH
- FISSION DISTRIBUTION

ESTIMATED CHANGE

CHANGE VERY SMALL AND ACCEPTABLE
CHANGES ARE INSIGNIFICANT

ESSENTIALLY UNCHANGED
ESSENTIALLY UNCHANGED

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SUMMARY EVALUATION OF LEAD, TiH SHIELD

NUCLEAR - EFFECTS NEGLIGIBLE

THERMAL - DESIGN ACCEPTABLE
OPERATING TEMPERATURES
TEMPERATURE GRADIENTS
PRESSURE DROP

<u>LEAD</u>	<u>TiH</u>
322°R	512°R
6°R	56°R
2.3 psi	1.7 psi

STRUCTURAL - DESIGN ACCEPTABLE
ULTIMATE STRENGTH
THERMAL STRESS

<u>LEAD</u>	<u>TiH</u>
2900 psi	10,000 psi
560 psi	8,000 psi

MECHANICAL - DESIGN ACCEPTABLE
LOW STRENGTH OF LEAD IS PERMISSIBLE SINCE IT IS
COMPLETELY SUPPORTED BY SHIELD STRUCTURE.
MATERIAL IS DUCTILE AND EASILY WORKED.

TiH IS BRITTLE BUT APPLICATION APPEARS FEASIBLE,
JUDGING FROM ACCEPTABLE THERMAL STRESSES
OBTAINED FOR ASSUMED GEOMETRY

SUMMARY - DESIGN APPEARS FEASIBLE

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INFLUENCE OF HIGH DENSITY SHIELD MATERIAL SUBSTITUTION ON STATE POINTS

- THE TOTAL HEAT DEPOSITED IN THE SHIELDS IS NOT SIGNIFICANTLY DIFFERENT WHEN BATH IS REPLACED BY HIGH DENSITY MATERIAL. THE TABLE BELOW SHOWS THE EFFECT OF A COMPLETE REPLACEMENT OF BATH BY $\text{BTiH}_{1.5}$:

	BATH	$\text{BTiH}_{1.5}$
EXTENSION SHIELD HEATING	0.21 MW	0.28 MW
PERIPHERAL SHIELD HEATING	0.75 MW	0.91 MW
	0.96 MW	1.19 MW
CENTRAL SHIELD HEATING	0.88 MW	1.10 MW

- THE CHANGE IN CYCLE STATE POINTS IS EXTREMELY SMALL DUE TO THE REPLACEMENT OF BATH BY HIGH DENSITY MATERIAL. THE TABLE BELOW SHOWS THE TEMPERATURE RISES FOR BATH AND FOR A FULL THICKNESS REPLACEMENT OF $\text{BTiH}_{1.5}$:

	BATH	$\text{BTiH}_{1.5}$
TEMPERATURE RISE* ACROSS EXTENSION SHIELD PLUS PERIPHERAL SHIELD	2.6 °R	3.2 °R
TEMPERATURE RISE* ACROSS CENTRAL SHIELD	2.3 °R	2.9 °R

THE PRESSURE DROPS CAN BE MADE ESSENTIALLY IDENTICAL BY DESIGN.

*BASED ON PARA-HYDROGEN PROPERTIES

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MECHANICAL DESIGN EVALUATION OF
A LEAD AND TITANIUM HYDRIDE SHIELD

STRUCTURAL CHARACTERISTICS OF LEAD

UNSUITABLE AS SELF-SUPPORTING STRUCTURAL ELEMENT

CHEMICAL AND PHYSICAL CHARACTERISTICS OF LEAD

LOW MELTING POINT (1080°R)

DUCTILE MATERIAL - STRESS NO PROBLEM

RADIATION STABILITY OF LEAD

NO PROBLEM

MANUFACTURING AND COST OF LEAD

LOW COST AND EASILY MANUFACTURED

NO MATERIAL DEVELOPMENT REQUIRED

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SCHEMATIC DRAWING OF REFERENCE DESIGN SHIELD
PER DRAWING 940J164

SECTION A-A
LEAD SHIELD SCHEMATIC - RADIAL FLOW

COMPARISON OF MATERIAL PROPERTIES

BATH VS BORATED TiH

PROPERTY * (ROOM TEMPERATURE)	BATH	TiH
T _s ULT psi	18,000	15,000
T _s Y psi	12,000	5000
Elongation %	2.75%	1/2 - 1%
C _s Y psi	13,500	-
E psi	9.555×10^6	13.0×10^6
ρ lb/in ³	.103	.137
Coef. Lin. Exp ln./ln. - °R	7.75×10^{-6}	6.0×10^{-6}
Coef. Th. Cond. BTU/FT/Hr-FT ² -°F	74.0	11.0
Specific Heat BTU/LB	.20	-
Melting Pt. °R	1691	-

*Note: THESE PROPERTIES ARE AVAILABLE AS $\bar{X}$ VALUES WITH STANDARD DEVIATION FOR BATH IN A RANGE OF TEMPERATURE 140 - 1080°R. (TME-1947, AUGUST 69)
THE PUBLISHED DATA ON TiH IS VERY LIMITED AND IS GENERALLY SHOWN AS 20° - 320°C. FOR THIS REASON ONLY ROOM TEMPERATURE DATA IS TABULATED FOR COMPARISON.

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SHIELD PROGRAM STATUS BASED ON 12-15-72 TEST LOGIC INDEPENDENT OF MATERIAL STATUS

SHIELD AVAILABLE FOR ASSEMBLY INTO R-1 NSS	05-07-72
PLACE HARDWARE PURCHASE ORDER	05-23-71
COMPLETE DETAIL DESIGN & ANALYSIS	01-17-71
START SHIELD LONG LEAD MATERIAL ERCO	09-13-70
COMPLETE MATERIAL PDS	06-10-70
START DETAIL DESIGN AND ANALYSIS	05-10-70
START SHIELD PRELIMINARY DESIGN	01-22-70
MAKE SHIELD MATERIAL DECISION	12-12-69

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MATERIAL DATA AND COMPONENT TEST CONSTRAINTS ON KEY ACTIVITIES WITH PRESENT FUNDING

<u>ACTIVITY</u>	<u>CONSTRAINT</u>	<u>STATUS</u>	
		<u>BATH</u>	<u>BTiH *</u>
COMPLETE CONCEPTUAL DESIGN 09-02-69	COMPLETE BATH MTL. "A" (LITERATURE) DATA	Complete	02-28-70
	COMPLETE MACHINING OF MTL.	Complete	Start 10-01-70
COMPLETE SHIELD PRELIMINARY DESIGN 05-10-70	COMPLETE MTL. "B" DATA	Complete (Less Fracture Mechanics & Rad.)	12-01-70 (Less Fracture Mechanics & Rad.)
	COMPLETE MTL. MACHINING TEST	01-13-70	Start 10-01-70
	COMPLETE CORROSION PARTS TEST	12-15-69	Start 10-01-70
PLACE R-1 MTL. PURCHASE ORDER 12-31-70	COMPLETE THERMAL SHOCK TEST	12-15-69	Start 10-01-70
	COMPLETE PDS SPEC.	06-10-70	03-01-71
COMPLETE SHIELD DETAIL DESIGN AND ANALYSIS 01-17-71	START MTL. ERCO	09-13-70	03-01-70 (24 wk. neg.)
	COMPLETE FLOW CHARAC- TERISTICS TEST	01-17-71	01-17-71
	COMPLETE MACHINING OF SPEC. GRADE MTL. TEST	01-17-71	10-01-71

* ASSUMES BATH MATERIALS DEVELOPMENT FUNDING REMAINING IN CY70 WOULD BE MADE AVAILABLE TO BTiH UPON MATERIALS DECISION 12-12-69.

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PROGRAMMATIC RISKS NECESSARY TO INSTALL BTIH SHIELD INTO R-1

1. PRELIMINARY AND DETAIL DESIGN IS ESSENTIALLY COMPLETED USING "A" DATA ONLY. INFORMATION ACQUIRED ON "B" DATA COULD REQUIRE REDESIGN.
2. MATERIAL ERCO MUST BE WRITTEN USING PRELIMINARY PDS SPEC. WRITTEN TO "A" DATA MATURITY.
3. MATERIAL PURCHASE ORDER MUST BE PLACED USING A PRELIMINARY PDS SPEC. "B" DATA OR FINAL SPEC. REQUIREMENTS COULD CAUSE MATERIAL RE-ORDER.
4. REACTOR HARDWARE MUST BE ORDERED USING MATERIAL BOUGHT TO PRE-LIMINARY MATERIAL SPEC. & UPGRADED AS REQUIRED AT RELEASE OF FINAL SPEC. NEW SPEC. REQUIREMENTS COULD DELAY HARDWARE MANUFACTURE.
5. DETAIL DESIGN MUST BE INITIATED AND CARRIED THROUGH TO NEAR COMPLETION WITHOUT RESULTS OF COMPONENT THERMAL SHOCK AND CORROSION OF PARTS TEST. NEGATIVE RESULTS IN THESE TESTS COULD REQUIRE REDESIGN.
6. ITEMS 1-5 ABOVE MAY BE IMPROVED ONLY AT THE EXPENSE OF OTHER COMPONENT DEVELOPMENT BASED ON THE PRESENT FUNDING SITUATION.

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CONCLUSIONS AND RECOMMENDATIONS

A HIGH DENSITY MATERIAL $\text{BTiH}_{1.5}$ CAN MEET THE ENGINE SPEC WITH A REDUCTION IN LENGTH OF ABOUT 4 INCHES WITH ESSENTIALLY EQUAL WEIGHT, RELATIVE TO REFERENCE DESIGN.

OTHER HIGH DENSITY MATERIALS, SUCH AS $\text{Pb} + \text{BTiH}_{1.5}$ AND $\text{BSS} + \text{BTiH}_{1.5}$, CANNOT MEET THE ENGINE SPEC WITHOUT AN INCREASE IN WEIGHT. THE REDUCTION IN LENGTH IS OF THE ORDER OF A FEW INCHES, RELATIVE TO REFERENCE DESIGN.

RECOMMENDATIONS

CONTINUE WITH BATH AS THE REFERENCE SHIELD MATERIAL FOR THE FOLLOWING REASONS:

1. THE REFERENCE DESIGN CAN, WITH SOME MODIFICATION, MEET THE PRESENT ENGINE SPECIFICATION.
2. THE BATH MATERIAL DATA SHOULD BE ABLE TO MEET SNPO REQUIREMENTS EXCEPT FOR IRRADIATION PROPERTIES AND FRACTURE MECHANICS.
3. INTRODUCTION OF A NEW MATERIAL WILL INCREASE RISKS AND REQUIRE SHIFTING OF MATERIALS EFFORT FROM MORE CRITICAL COMPONENTS.

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TASK 24.W - WANL STUDY OF REFLECTOR FORWARD END SHIELD

1. INTRODUCTION
2. DESCRIPTION OF CONCEPT
3. RADIATION AND SHIELDING ANALYSIS OF MERIDIAN
RING DATA
4. SUMMARY

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W58835-1

TASK 20.1W - WANL ESTIMATE OF UNCERTAINTIES
IN PVARA SOURCE TERMS

1. INTRODUCTION
2. DESCRIPTION OF CRAM MODEL
3. KEY SOURCES BY REGION IN THE PVARA
4. IMPORTANCE OF KEY SOURCES ON THE MERIDIAN RING
5. DESCRIPTION OF PAX/G1 AND R-1 PARAMETRIC SHIELD EXPERIMENTS
6. COMPARISON OF CALCULATED AND MEASURED DATA INTERNAL TO THE REACTOR
7. UNCERTAINTY FACTORS FOR PVARA SOURCES
8. EFFECT OF UNCERTAINTY FACTORS ON LEAKAGE

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DESCRIPTION OF COMMON RADIATION ANALYSIS MODEL (CRAM)

THE CRAM CONTAINS THE FOLLOWING DATA PERTINENT TO THE NSS:

- GEOMETRY MODEL (APPROXIMATELY 30 REGIONS)
- MATERIAL COMPOSITIONS BY REGION
- RELATIVE RADIAL AND RELATIVE AXIAL FISSION DISTRIBUTION
IN THE CORE
- RELATIVE RADIAL AND RELATIVE AXIAL DISTRIBUTION OF THE
PHOTON SOURCE IN EACH REGION
- TABULATED 13 GROUP PHOTON SOURCE STRENGTH BY REGION

KEY SOURCES OF GAMMA RADIATION IN THE PVARA *

<u>COMPONENT</u>	<u>SOURCE STRENGTH (MEV/SEC)</u>
REACTOR CORE	8.85×10^{20}
REFLECTOR	2.12×10^{19}
SUPPORT PLATE	7.07×10^{18}
SHIELD	1.96×10^{18}
BORAL LINER	2.41×10^{18}
AFT END HARDWARE	3.38×10^{18}
OTHER	<u>9.50×10^{18}</u>
TOTAL NSS	9.31×10^{20}
PRESSURE VESSEL	<u>1.57×10^{17}</u>
TOTAL PVARA	9.31×10^{20}

* BASED ON R-1 REACTOR 939J685-E

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CONTRIBUTION OF NSS COMPONENTS TO GAMMA RAY DOSE RATE
AT SEVERAL LOCATIONS ON A 30 FOOT MERIDIAN RING

REACTOR COMPONENT	GAMMA DOSE RATE IN PERCENT OF TOTAL AT EACH LOCATION		
	0° (REACTOR CENTERLINE)	30°	90° (REACTOR MIDPLANE)
CORE	39.6	65.6	89.5
REFLECTOR	10.0	21.9	6.0
SUPPORT PLATE	28.9	1.1	0.7
SHIELD	9.0	2.3	0.3
BORAL LINER	0.3	7.3	2.0
OTHER	12.2	1.8	1.5
TOTAL	100.0	100.0	100.0
DOSE RATE RADS-(CARBON)/HR	1.1 (5)	1.8 (6)	1.3 (7)

NOTE: NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

DESCRIPTION OF PAX/G1 AND R-1 PARAMETRIC EXPERIMENTS

- RECENT PAX/G1 EXPERIMENTS WERE PERFORMED TO SUPPORT RADIATION AND SHIELDING ANALYSIS ON A MOCKUP OF A 60° SECTOR OF THE R-1 CORE AND REFLECTOR (939J685A)
- MEASUREMENTS WERE OBTAINED IN THE CORE, LATERAL SUPPORT, AND REFLECTOR REGIONS OF THE REACTOR AS REPORTED IN TME-1914
- THE FOLLOWING TYPES OF MEASUREMENTS WERE PERFORMED:
 1. SULFUR TO OBTAIN THE FAST NEUTRON FLUX ($E > 2.9$ Mev),
 2. U-238 TO OBTAIN THE FAST NEUTRON FLUX ($E > 1.5$ Mev),
 3. BARE AND CADMIUM COVERED DYSPROSIUM TO OBTAIN THERMAL AND INTERMEDIATE ENERGY NEUTRON FLUXES, AND
 4. CaF_2 TLD'S TO OBTAIN THE GAMMA RAY DOSE RATES
- SIMILAR MEASURED DATA WERE OBTAINED AT PAX IN A MOCKUP OF THE SUPPORT PLATE AND BATH SHIELD (TME-1852)
- SELECTED COMPARISONS OF THESE EXPERIMENTAL AND CALCULATED DATA WERE EMPLOYED TO DETERMINE UNCERTAINTY FACTORS FOR THE NSS SOURCES

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PROCEDURE EMPLOYED TO DETERMINE EQUIVALENT FLUX DATA

- A FOIL IS IRRADIATED AT CONSTANT POWER LEVEL FOR A PERIOD OF TIME IN A NEUTRON FLUX, ϕ (E), TO PRODUCE A RADIOACTIVE NUCLIDE (s) THAT EMITS β OR γ RAYS.
- THE FOIL IS PLACED IN A NUCLEAR COUNTER THAT MEASURES THE β OR γ DECAY RATE OF THE RADIOACTIVE NUCLIDE (s).
- FROM THE MEASURED BETA OR GAMMA DECAY RATE, THE RATE OF PRODUCTION OF THE RADIOACTIVE NUCLIDE (s) (COMMONLY REFERRED TO AS SATURATED ACTIVITY) IS DETERMINED FROM BASIC PRODUCTION AND DECAY RELATIONSHIPS. THIS SATURATED ACTIVITY IS DENOTED AS E_a .
- DOT DIRECTLY CALCULATES THE SATURATED ACTIVITY, DENOTED AS C_a .
- BOTH SETS OF DATA ARE REDUCED TO AN EQUIVALENT FLUX, $\bar{\phi}$ (ΔE) IN A GIVEN ENERGY INTERVAL ΔE USING THE SAME SPECTRAL WEIGHTED CROSS SECTION $\bar{\sigma}$ BY USING THE RELATIONSHIP:

$$\begin{aligned} \text{EXPERIMENTALLY MEASURED FLUX} &\equiv \bar{\phi}_E(\Delta E) = \frac{E_a}{\bar{\sigma}} \\ \text{DOT CALCULATED FLUX} &\equiv \bar{\phi}_c(\Delta E) = \frac{C_a}{\bar{\sigma}} \end{aligned}$$

COMPARISON OF CALCULATED AND EXPERIMENTAL
RELATIVE RADIAL FISSION FLUX

COMPARISON OF CALCULATED AND EXPERIMENTAL
RELATIVE AXIAL FISSION FLUX

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL
U-238 ($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF
RADIAL POSITION

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
(E>1.5 Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

Data obtained using GAMBIT 16 group neutron cross section data (11 fast groups above 1.86 ev, and 5 thermal groups below 1.86 ev) derived from WANL evaluated 155 group neutron cross section data (62 fast groups above 1.86 ev and 93 thermal groups below 1.86 ev) spectrally reweighted over predicted neutron spectra in NSS regions via the ANISN code.

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

BARE DYSPROSIUM NEUTRON FLUX (N/CM²-SEC-WATT)

Data obtained using GAMBIT 16 group neutron cross section data (11 fast groups above 1.86 ev, and 5 thermal groups below 1.86 ev) derived from WANL evaluated 155 group neutron cross section data (62 fast groups above 1.86 ev and 93 thermal groups below 1.86 ev) spectrally reweighted over predicted neutron spectra in NSS regions via the ANISN code.

— AT CORE MIDPLANE
DOT CALCULATED DATA
◇ EXPERIMENTAL DATA
(BARE DYSPROSIUM)

CORE

LATERAL SUPPORT

REFLECTOR

PRESSURE VESSEL

RADIUS (CM)

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE
DYSPROSIUM ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

Data obtained using GAMBIT 16 group neutron cross section data (11 fast groups above 1.86 ev, and 5 thermal groups below 1.86 ev) derived from WANL evaluated 155 group neutron cross section data (62 fast groups above 1.86 ev and 93 thermal groups below 1.86 ev) spectrally reweighted over predicted neutron spectra in NSS regions via the ANISN code.

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL
DYSPROSIUM ($E < 0.4$ eV) THERMAL NEUTRON
FLUX AS A FUNCTION OF RADIAL POSITION

CONFIDENTIAL
RESTRICTED DATA
ATOMIC ENERGY ACT OF 1954 GP-1

CONFIDENTIAL
RESTRICTED DATA
ATOMIC ENERGY ACT OF 1954 GP-1
SEMILOGARITHMIC
CAF₂ TLD GAMMA ABSORBED DOSE RATE (RADS-C/HR-WATT)

CONFIDENTIAL
RESTRICTED DATA
ATOMIC ENERGY ACT OF 1954 GP-1

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF_2 TLD)
GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

CaF_2 TLD GAMMA ABSORBED DOSE RATE (RADS-CARBON/HR-WATT)

AXIAL DISTANCE FROM CORE INLET - (CM)

CONFIDENTIAL
RESTRICTED DATA

EL & ESSER
3/4/54
UNCLASSIFIED

CONFIDENTIAL
RESTRICTED DATA
ATOMIC ENERGY ACT OF 1954 Gp 1

GAMMA RAY DOSE RATE ON THE REACTOR MIDPLANE DUE TO ALL PHOTON SOURCES AND CORE PHOTON SOURCES ONLY AS A FUNCTION OF RADIUS

(CALCULATED BY DOT)

GAMMA RAY DOSE RATE (RADS-C/HR-W)

RADIUS (CM)

DOSE RATE DUE TO ALL PHOTON SOURCES

DOSE RATE DUE TO CORE PHOTON SOURCES ONLY

COMPARISON OF EXPERIMENTAL AND CALCULATED
 U-238 FISSION RATE (FAST NEUTRON FLUX, $E > 1.5$
 MEV) VERSUS AXIAL POSITION ON THE REACTOR
 CENTERLINE, CONFIGURATION 4B

COMPARISON OF EXPERIMENTAL AND CALCULATED
SATURATED BARE GOLD-197 ACTIVITY (THERMAL
NEUTRON FLUX) VERSUS AXIAL POSITION ON THE
REACTOR CENTERLINE, CONFIGURATION 4B

COMPARISON OF EXPERIMENTAL AND CALCULATED GAMMA RAY
ABSORBED DOSE RATE VERSUS AXIAL POSITION ON THE REACTOR
CENTERLINE, CONFIGURATION 4B

GAMMA RAY ABSORBED DOSE RATE (RADS(C)/HR-WATT)

CALCULATED FRACTION OF NEUTRON ABSORPTIONS BY GROUP IN VARIOUS NSS REGIONS

<u>REGION</u>	<u>NEUTRON GROUP</u>			<u>TOTAL</u>
	<u>FAST</u> <u>E > 1.35 MEV</u>	<u>INTERMEDIATE</u> <u>0.4 EV < E < 1.35 MEV</u>	<u>THERMAL</u> <u>E < 0.4 EV</u>	
AFT END SUPPORT	0.0126	0.9671	0.0203	1.000
SUPPORT PLATE	0.00744	0.278	0.716	1.000
SHIELD	0.00119	0.720	0.280	1.000
REFLECTOR	0.00306	0.186	0.811	1.000
SIDE BORAL LINER	0.000394	0.452	0.549	1.000
CORE	0.0225	0.955	0.0231	1.000

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RATIO OF MEASURED TO CALCULATED FOIL ACTIVATION IN VARIOUS REGIONS

REGION	FAST ⁽¹⁾ (E > 1.5 MEV)	EPICADMIUM ⁽²⁾ (E > 0.4 EV)	SUBCADMIUM ⁽³⁾ (E < 0.4 EV)
AFT END SUPPORT	0.60	2.5	5.3
SUPPORT PLATE	0.93	1.37	0.63
SHIELD	0.76	0.61	0.019
REFLECTOR	0.56	1.07	1.09
SIDE BORAL LINER	0.1	1.75	1.38
CORE	0.98	1.28	3.68

(1) U^{238} DATA

(2) CADMIUM COVERED DYSPROSIUM, EXCEPT IN THE SHIELD WHERE CADMIUM COVERED GOLD WAS USED.

(3) BARE MINUS Cd COVERED DATA.

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NSS SOURCE UNCERTAINTY FACTORS FOR KEY REGIONS

COMPONENT	CALCULATED SOURCE STRENGTH (MEV/SEC)	UNCERTAINTY* FACTOR	PERCENT STANDARD DEVIATION IN THE EXPERIMENTAL DATA**
REACTOR CORE	8.85×10^{20}	1.07	+ 1.0 % —
REFLECTOR	2.12×10^{19}	1.08	+ 14.1 —
SUPPORT PLATE	7.07×10^{18}	0.84	+ 35.4 —
SHIELD	1.96×10^{18}	0.44	+ 186.0 —
BORAL LINER	2.41×10^{18}	1.55	+ 14.8 —
AFT END HARDWARE	3.38×10^{18}	2.53	+ 11.2 —
OTHER	9.50×10^{18}	NOT KNOWN	NOT KNOWN

* FACTOR TO BE USED TO ADJUST CALCULATED DATA TO NOMINAL MEASUREMENT
WEIGHTED ACCORDING TO ABSORPTIONS BY NEUTRON GROUP, EXCEPT IN THE CORE

** UNCERTAINTY IN EXPERIMENTAL DATA WEIGHTED ACCORDING TO ABSORPTIONS BY NEUTRON
GROUP, EXCEPT IN THE CORE

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RELATIVE EFFECT OF UNCERTAINTY FACTOR ON
MERIDIAN RING DATA AT 30 FEET

POLAR ANGLE = 0°

<u>SOURCE REGION</u>	<u>CALCULATED</u>	<u>CALCULATED X UNCERTAINTY FACTOR</u>
CORE	39.6	42.4
REFLECTOR	10.0	10.8
SUPPORT PLATE	28.9	24.2
SHIELD	9.0	3.98
BORAL LINER	0.3	0.47
OTHER	<u>12.2</u>	<u>12.2</u>
TOTAL	100.0	94.05 $\pm$ 11.4

POLAR ANGLE = 30°

<u>SOURCE REGION</u>	<u>CALCULATED</u>	<u>CALCULATED X UNCERTAINTY FACTOR</u>
CORE	65.6	70.2
REFLECTOR	21.9	23.7
SUPPORT PLATE	1.1	0.92
SHIELD	2.3	1.02
BORAL LINER	7.3	11.3
OTHER	<u>1.8</u>	<u>1.8</u>
TOTAL	100.0	108.94 $\pm$ 3.8

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SUMMARY

- DETERMINATION OF UNCERTAINTY FACTORS APPLICABLE TO CALCULATED RADIATION SOURCES IS EXTREMELY DIFFICULT
- NO DIRECT MEASUREMENT OF THE PHOTON SOURCE IS AVAILABLE
- UNCERTAINTY FACTORS DETERMINED FROM THIS STUDY WERE BASED ON AN EVALUATION OF AVAILABLE MEASURED VERSUS CALCULATED ACTIVATION NEUTRON DATA
- HOWEVER, NEUTRON ACTIVATION DATA ARE NOT AVAILABLE FOR MOST MATERIALS OF INTEREST IN THE NSS
- VERIFICATION OF THE UNCERTAINTY FACTORS RESULTING FROM THIS STUDY REQUIRE AN EXPANDED EXPERIMENTAL PROGRAM IN WHICH APPROPRIATE MATERIAL ARE ACTIVATED IN THE REACTOR

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TASK 21.1 - WANL COMPUTATION OF BORAL LINER EFFECTIVENESS

1. INTRODUCTION
2. DESCRIPTION OF ANALYTICAL PROCEDURE
3. MULTIGROUP APPROXIMATIONS USED
4. DESCRIPTION OF BASIC CROSS SECTION DATA
5. DESCRIPTION OF CRAM MODEL AND VARIATIONS
6. COMPARISONS OF NEUTRON LEAKAGE SPECTRUM FROM PRESSURE VESSEL WITH AND WITHOUT BORAL LINER
7. COMPARISON OF NEUTRON INDUCED GAMMA SOURCES IN AI 6061 AND HAST-C AT RADIAL MIDPLANE
8. EFFECT OF NEUTRON CHANNELING THROUGH B_4C CHUNK DISPERSION IN BORAL ON NEUTRON INDUCED GAMMA SOURCE
9. COMPARISON OF EXPERIMENTAL THERMAL AND FAST NEUTRON RESPONSE RATES EXTERNAL TO PAX SERIES OF REACTORS
10. SUMMARY AND CONCLUSIONS

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METHOD OF ANALYSIS

STUDY PARAMETERS AND ASSUMPTIONS

- ALL ANALYSES PERFORMED WITH MAY '69 CRAM MODEL WITH MODIFICATIONS TO OBTAIN RELATIVE EFFECTS. DIMENSIONAL AND MODEL MESH WERE CONSISTANT FOR ALL ANALYSES
- NO CONTROL DRUM REPRESENTATION IN REFLECTOR
- BORAL LINER EXPLICITLY DEFINED IN MODEL WITH CLAD AND MEAT IN BORAL SANDWICH REPRESENTED BY SEVEN (7) MESH INTERNALS
- MODEL WITHOUT BORAL LINER WAS ACHIEVED BY PLACING VOID IN BORAL LINER POSITION
- ALL NEUTRON INDUCED GAMMA SOURCE DATA IS BASED ON NEUTRON LEAKAGE SPECTRA FROM PRESSURE VESSEL AT THE RADIAL MIDPLANE

MULTIGROUP APPROXIMATIONS

(Numbers on Group Break points are Energies)

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DESCRIPTION OF BASIC CROSS SECTION DATA

AGC-H/R

MULTIGROUP TRANSPORT CORRECTED CROSS SECTION DATA WEIGHED OVER INFINITE MEDIA SPECTRUM. THE ORIGINAL HANSEN AND ROACH LIBRARY (LASL-) HAS BEEN UPDATED BY AGC TO PROVIDE A LIBRARY CONTAINING THE MAJORITY OF ELEMENTS IN THE NERVA ENGINE.

WANL-WISDM(16)
-WISDM(52)
-H/R

MULTIGROUP TRANSPORT CORRECTED CROSS SECTION DATA WEIGHED OVER NEUTRON FLUX AND CURRENT SPECTRUM REPRESENTATIVE OF THE REGION IN WHICH THE MATERIAL EXISTS IN THE MODEL. TRANSPORT CORRECTION IS PERFORMED FOR EACH REGION. BASIC CROSS SECTION DATA (FINE GROUP AND RESONANCE PARAMETERS) HAVE BEEN EVALUATED BY WANL FROM ENDF/B AND OTHER SOURCES AND DATA EXISTS ON DATA FILES FOR USE IN GAMBIT. GAMBIT GENERATES THESE MULTIGROUP DATA FROM RESONANCE PARAMETERS AND SPECTRAL REWEIGHING TO THE CHOSEN ENERGY GROUP STRUCTURE.

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$\frac{J(\text{WITH BORAL})}{J(\text{WITHOUT BORAL})}$, RATIO OF NEUTRON LEAKAGE FROM PRESSURE VESSEL

COMPARISON OF BORAL ATTENUATION EFFECTIVENESS ON NEUTRON
INDUCED GAMMA SOURCE FOR VARIOUS MULTIGROUP APPROXIMATIONS

(GAMMA SOURCE IN 1 CM³ OF HAST-C, $\frac{\text{MEV}}{\text{CM}^3\text{-SEC}}$)

		<u>W/O BORAL</u>	<u>W/BORAL</u>		<u>RATIO*</u>	
			<u>HOMO- GENEOUS</u>	<u>INHOMO- GENEOUS**</u>	<u>HOMO- GENEOUS</u>	<u>INHOMO- GENEOUS</u>
WANL WISDM (52)	C ***	2.65 (+14)	6.49 (+13)	1.12 (+14)	0.25	0.42
	I	5.11 (+14)	4.95 (+12)	4.93 (+12)	0.97	0.96
	T	2.70 (+14)	6.98 (+13)	1.17 (+14)	0.26	0.43
AGC H/R (16)	C	5.93 (+14)	1.95 (+14)	---	0.32	--
	I	---	---	---	--	--
	T	5.93 (+14)	1.95 (+14)	---	0.32	--
WANL H/R (16)	C	2.85 (+14)	6.00 (+13)	---	0.21	--
	I	5.07 (+12)	4.90 (+12)	---	0.97	--
	T	2.91 (+14)	6.49 (+13)	---	0.22	--
WANL WISDM (16)	C	3.06 (+14)	6.13 (+13)	---	0.20	--
	I	5.07 (+12)	4.90 (+12)	---	0.97	--
	T	3.11 (+14)	6.63 (+13)	---	0.21	--

* GAMMA SOURCE WITH BORAL LINER
* GAMMA SOURCE WITHOUT BORAL LINER

**INHOMOGENEOUS CALCULATION CONSIDERS B₄C PARTICLE DISTRIBUTION IN BORAL

*** C - RADIATIVE CAPTURE, I - INELASTIC SCATTER, T - TOTAL

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COMPARISON OF EFFECT OF MULTIGROUP APPROXIMATION AND NEUTRON CROSS
SECTION DATA ON NEUTRON INDUCED GAMMA SOURCE (CAPTURE ONLY) BY ELEMENT

(MEV/SEC-CM³ IN HAST-C)

HAST-C		AGC H/R (16)	WANL H/R (16)	WANL H/R (16) WITH AGC H/R (16) NEUTRON LEAKAGE DATA
Cr	w/*	2.42 (+12)	1.68 (+12)	1.52 (+12)
	w/o	3.25 (+13)	2.30 (+13)	2.32 (+13)
Mn	w/	**	1.49 (+12)	1.30 (+12)
	w/o	**	6.64 (+12)	6.53 (+12)
Fe	w/	2.60 (+11)	3.71 (+11)	3.42 (+11)
	w/o	5.69 (+12)	4.41 (+12)	4.45 (+12)
Ni	w/	1.59 (+13)	8.87 (+12)	8.88 (+12)
	w/o	1.71 (+14)	1.23 (+14)	1.25 (+14)
Mo	w/	2.67 (+13)	1.73 (+13)	1.56 (+13)
	w/o	4.65 (+13)	3.58 (+13)	3.32 (+13)
W	w/	1.31 (+14)	3.03 (+13)	2.726 (+13)
	w/o	2.76 (+14)	9.23 (+13)	8.69 (+13)
Co	w/	1.83 (+13)	***	***
	w/o	6.13 (+13)	***	***

* w/ - WITH BORAL, w/o - WITHOUT BORAL

** AGC H/R DATA DID NOT INCLUDE Mn AT TIME OF ANALYSIS

*** WANL GAMBIT FOR WISDM DID NOT INCLUDE COMPLETE COBALT DATA AT TIME OF ANALYSIS
(NEW EVALUATED DATA IS NOW AVAILABLE)

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COMPARISON OF BORAL ATTENUATION EFFECTIVENESS ON NEUTRON INDUCED GAMMA SOURCE
IN THE PUMP DISCHARGE LINE (Al6061 WITH LH₂)

(GAMMA SOURCE, MEV/SEC IN SECTION OF Al6061 PDL AT RADIAL MIDPLANE)

	WITHOUT BORAL LINER	WITH BORAL LINER		RATIO *	
		Homogeneous	Inhomogeneous	Homogeneous	Inhomogeneous **
Al6061	2.73 (+15)	1.54 (+15)	1.88 (+15)	0.56	0.69
LH ₂	7.57 (+15)	5.10 (+15)	5.95 (+15)	0.67	0.79
TOTAL	10.5 (+15)	6.64 (+15)	7.83 (+15)	0.63	0.75

NOTE: DATA BASED ON WANL WISDM (52) DATA FROM R-θ DOT

* RATIO = $\frac{\text{MEV/SEC WITH BORAL LINER}}{\text{MEV/SEC WITHOUT BORAL LINER}}$

** INHOMOGENEOUS CALCULATION INCLUDES EFFECT OF B₄C PARTICLE SIZE DISTRIBUTION IN BORAL MEAT OF LINER.

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COMPARISON OF BORAL ATTENUATION EFFECTIVENESS ON NEUTRON
INDUCED GAMMA SOURCE FOR VARIOUS MULTIGROUP APPROXIMATIONS

(GAMMA SOURCE IN 1 CM³ OF Al 6061, $\frac{\text{MEV}}{\text{CM}^3\text{-SEC}}$)

		<u>W/O BORAL</u>	<u>W/BORAL</u>		<u>RATIO*</u>	
			<u>HOMO- GENEOUS</u>	<u>INHOMO- GENEOUS**</u>	<u>HOMO- GENEOUS</u>	<u>INHOMO- GENEOUS</u>
WANL WISDM (52)	C***	5.76 (+12)	7.55 (+11)	1.76 (+12)	0.13	0.31
	I	1.09 (+12)	1.06 (+12)	1.05 (+12)	0.97	0.96
	T	6.85 (+12)	1.82 (+12)	2.85 (+12)	0.26	0.42
AGC H/R (16)	C	8.51 (+12)	5.91 (+11)	---	0.069	--
	I	---	---	---	---	--
	T	8.51 (+12)	5.91 (+11)	---	0.069	--
WANL H/R (16)	C	6.43 (+12)	5.79 (+11)	---	0.090	--
	I	1.05 (+12)	1.02 (+12)	---	0.97	--
	T	7.48 (+12)	1.60 (+12)	---	0.21	--
WANL WISDM (16)	C	6.88 (+12)	5.82 (+11)	---	0.084	--
	I	1.05 (+12)	1.02 (+12)	---	0.97	--
	T	7.93 (+12)	1.60 (+12)	---	0.20	--

* GAMMA SOURCE WITH BORAL LINER
* GAMMA SOURCE WITHOUT BORAL LINER

** INHOMOGENEOUS CALCULATION CONSIDERS B₄C PARTICLE DISTRIBUTION IN BORAL

*** C - RADIATIVE CAPTURE, I - INELASTIC SCATTER, T - TOTAL

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COMPARISON OF EFFECT OF MULTIGROUP APPROXIMATIONS AND
NEUTRON CROSS SECTIONS ON NEUTRON INDUCED SOURCES

		(MEV/SEC-CM ³ IN HAST-C)		(MEV/SEC-CM ³ IN Al 6061)	
		<u>W/BORAL</u>	<u>W/O BORAL</u>	<u>W/BORAL</u>	<u>W/O BORAL</u>
WANL WISDM (52)	C	6.49 (+13)	2.65 (+14)	7.55 (+11)	5.76 (+12)
	I	4.95 (+12)	5.11 (+12)	1.06 (+12)	1.09 (+12)
	T	6.98 (+13)	2.70 (+14)	1.82 (+12)	6.85 (+12)
AGC H/R (16)	C	1.95 (+14)	5.93 (+14)	5.91 (+11)	8.51 (+12)
	I	---	---	---	---
	T	1.95 (+14)	5.93 (+14)	5.91 (+11)	8.51 (+12)
WANL H/R (16)	C	6.00 (+13)	2.85 (+14)	5.79 (+11)	6.43 (+12)
	I	4.91 (+12)	5.07 (+12)	1.02 (+12)	1.05 (+12)
	T	6.49 (+13)	2.91 (+14)	1.60 (+12)	7.48 (+12)
WANL H/R (16) W/AGC LEAKAGE DATA	C	5.49 (+13)	2.80 (+14)	5.85 (+11)	6.54 (+12)
	I	8.34 (+12)	8.62 (+12)	1.56 (+12)	1.61 (+12)
	T	6.33 (+13)	2.88 (+14)	2.14 (+12)	8.15 (+12)
WANL WISDM (16)	C	6.13 (+13)	3.06 (+14)	5.82 (+11)	6.88 (+12)
	I	4.91 (+12)	5.07 (+12)	1.02 (+12)	1.05 (+12)
	T	6.63 (+13)	3.11 (+14)	1.59 (+12)	7.93 (+12)

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COMPARISON OF EXPERIMENTALLY MEASURED BORAL ATTENUATION
EFFECTIVENESS AND CALCULATED PREDICTIONS
(WANL WISDM 52 GROUP DATA)

RATIO: $\frac{\text{ABSORPTIONS WITHOUT BORAL}}{\text{ABSORPTIONS WITH BORAL}}$

	EXPERIMENTAL*	CALCULATED (52 GROUP WISDM)	
		<u>HOMOGENEOUS</u>	<u>INHOMOGENEOUS**</u>
EPICADMIUM 1/V ABSORBER	2.2 ± 0.3	2.80	1.64
SUBCADMIUM 1/V ABSORBER	44.2 ± 11.2	90.0	7.4
BARE 1/V ABSORBER	10.4 ± 1.5	21.4	5.4

* EXPERIMENTAL DATA ARE PAX-E5 MEASUREMENTS IN KIWI REFLECTOR SECTORS
(SEE WANL-TME-1852)

** TRANSMISSION OF NEUTRONS THROUGH 1/8" BORAL CALCULATED ON BASIS
OF BORAL COMPOSITION OF 35 V/O B₄C IN MEAT WITH 20-100 MESH PARTICLE
SIZE DISTRIBUTION OF B₄C PARTICLES IN Al MATRIX.

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COMPARISON OF EXPERIMENTALLY MEASURED BORAL ATTENUATION EFFECTIVENESS AND CALCULATED PREDICTIONS

$$\text{RATIO: } \frac{\text{FOIL ACTIVITY WITHOUT BORAL}}{\text{FOIL ACTIVITY WITH BORAL}}$$

FOIL	EXPERIMENTAL*	CALCULATED (52 GROUP WISDM)	
		HOMOGENEOUS	INHOMOGENEOUS**
EPICADMIUM Dy ¹⁶⁴	3.2 ± 0.5	4.7	2.3
	3.3 ± 0.5		
	2.9 ± 0.4		
SUBCADMIUM Dy ¹⁶⁴	45 ± 6	62.0	7.8
	38 ± 5		
	26 ± 4		
BARE Dy ¹⁶⁴	75 ± 8	113.0	9.0
	56 ± 5		
	35 ± 3		

* EXPERIMENTAL DATA ARE PAX-E5 MEASUREMENTS IN NRX-A6 AND KIWI REFLECTOR SECTORS
(SEE WANL-TME-1852).

** TRANSMISSION OF NEUTRONS THROUGH 1/8" BORAL LINER CALCULATED ON BASIS OF BORAL COM-
POSITION OF 35 V/O B₄C IN MEAT WITH 20-100 MESH PARTICLE SIZE DISTRIBUTION OF B₄C PARTICLES
IN Al MATRIX.

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SUMMARY AND CONCLUSIONS

- ANALYSES USING CONSISTENT MODELS AND TECHNIQUES TO COMPUTE NEUTRON LEAKAGE FROM THE NSS GIVE CONSISTENT RESULTS EXCEPT FOR NEUTRON INDUCED GAMMA SOURCES IN HIGH Z MATERIALS.
- DIFFERENCES IN NEUTRON RADIATIVE CAPTURE CROSS SECTIONS FOR HIGH Z MATERIALS ARE MAJOR CONTRIBUTORS TO DIFFERENCES IN WANL AND AGC GAMMA SOURCE RESULTS.
- AT AGC'S REQUEST, WANL WILL SUPPLY A 36 ELEMENT NEUTRON CROSS SECTION LIBRARY IN 52 GROUPS (32 FAST GROUPS ABOVE 1.86 EV AND 20 THERMAL GROUPS BELOW 1.86 EV). DATA WILL INCLUDE LATEST WANL EVALUATED DATA (ENDF/B AND OTHER MORE CURRENT SOURCES) IN TRANSPORT CORRECTED AND P_2 (P_3 FOR 16 ELEMENTS) CROSS SECTION MATRICES.
- ATTENUATION EFFECTIVENESS OF BORAL LINER IN REDUCING ENGINE RADIATION SOURCES ARE:
 - A. A FACTOR OF ~4.0 (2.5 - 5.0) REDUCTION IN GAMMA SOURCE FOR TURBINE INLET OR EXHAUST (HASTELLOY) LINE COMPONENTS.
 - B. A FACTOR OF ~1.4 (1.2 - 1.5) REDUCTION IN GAMMA SOURCE FOR PUMP DISCHARGE LINE (Al 6060 WITH LIQUID HYDROGEN).

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TASK 22.1W - WANL REQUIREMENTS FOR NSS COMPONENTS
ABOVE THE INTERNAL SHIELD

1. INTRODUCTION
2. IN-FLIGHT 10 HOUR OPERATING ENVIRONMENT
AT THE CDA AND SSCV
3. AFTER SHUTDOWN ENVIRONMENT AT THE CDA AND
SSCV FOR VARIOUS REACTOR RUNS
4. CONCLUSIONS

POTENTIOMETER (SIMILAR MATERIALS TO THOSE USED IN MARKITE POTS FOR XE, NRX-A6
ACTUATOR SIDE-BY-SIDE, AND PLUM BROOK ACTUATOR TESTS

MATERIAL	APPLICATION	**TEMPERATURE ENVIRONMENT, °R	
		NOMINAL	3σ
302 SS	END BELLS	320	570
NITRALLOY OR INCONEL 718	SHAFT	360	570
GOLD	SLIP RINGS	290	380
RHODIUM	CONTACT WHISKER	270	300
CARBON SILVER	RESISTANCE TRACK	260	290
PLASTIC (MARKITE)	TRACK BASE	270	290
BERYLLIUM COPPER	WIPER ARM	260	280
PRECIOUS METAL	WIPER CONTACT FINGERS	300	360
ALLEGHENY LUDLUM RELAY 5 (SOFT IRON)	MAGNET FIELD PIECES	500	860
ALLEGHENY LUDLUM RELAY 5 (SOFT IRON)	HAMMER	280	340
302 SS	HAMMER SPRING	322	470
ALNICO V	TRIGGER MAGNET	390	530
COPPER WITH POLYIMIDE INSULATION	TRIGGER COIL	260	280
PLASTIC	COIL BOBBIN	290	380
NITRALLOY OR 302 SS WITH TUFTRIDE	RAM	390	680
NITRALLOY OR 302 SS WITH TUFTRIDE	DRIVE SHAFT ENDS	470	900
440 C SS	LATCH BALLS	390	680
302 SS WITH TUFTRIDE	LATCH BALL RETAINER RING	300	380
302 SS	LATCH SPRING	300	400
NITRALLOY	HELICAL CAM	360	570
304 CONDITION A	DRIVE SHAFT TUBE	349	510
302 SS	MOD SPRING	430	790
NITRALLOY	CAM FOLLOWER PIN	410	700
302 SS	MARMAN CLAMP	540	600
CARBON SPRING STEEL SAE 1065-1090	SNAP RINGS & BELLEVILLE WASHERS	330	400
300 SERIES SS	SCREWS	310	450

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CONTROL DRUM ACTUATOR
PRESSURE VESSEL MOUNTED - UNSHIELDED ENVIRONMENT

UNSHIELDED RADIATION ENVIRONMENT -

GAMMA - 1.1×10^{11} , ERGS/GM-C; FAST - 3.6×10^{17} , N/CM²; THERMAL - 4×10^{17} , N/CM²

MATERIAL	APPLICATION	TEMPERATURE ENVIRONMENT, °R	
		NOMINAL	3σ
AL 7075-T6	HOUSING	410	600
	HOUSING COVER	390	680
	CONNECTOR		
302 SS	SHELL	520	770
302 SS	PIN	540	610
PSC DUROCK	INSULATOR	410	700
303 SS	SOCKET	450	700
INCONEL	SPRING	570	820
COPPER W/POLYIMIDE INSULATION	LEAD WIRE	300	400
ALUMINUM 7075-T6 W/LEAD PLATE	SEALS	250	260
INCONEL 718 OR 302 SS W/TUFTRIDE LOCALLY ON SPLINE TEETH	FLEXSPLINE	260	270
	MOTOR		
302SS	HOUSING	320	570
302 SS	END BELL	320	570
ALLEGHENY LUDLUM DYNAMO M-27	STATOR LAMINATIONS	290	360
ALLEGHENY LUDLUM RELAY NO. 5	ROTOR CORE	390	680
ALNICE V OR SAMARIUM-COBALT	ROTOR MAGNETS	390	680
302 SS	SHAFT	500	1000
440 C SS, GOLD PLATED	BEARINGS	390	680
COMPOSITION FILLED W/MOLYDISELINIDE	BEARING RETAINERS	270	300
440 C SS	WAVE GENERATOR	290	360
COMPOSITION FILLED W/MOLYDISELINIDE	BALL RETAINER	270	320
NITRALLOY	OLDHAM COUPLING	390	680

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RADIATION EFFECTS UNSHIELDED ENVIRONMENT

1. THE USE OF METALS AS SPRINGS: RADIATION COULD CAUSE STRESS RELAXATION OF 10 TO 20 PERCENT, DEPENDING ON TEMPERATURE AND STRESS LEVELS.
2. ELECTRICAL CONDUCTORS: FAST NEUTRON IRRADIATION INCREASES THE ELECTRICAL RESISTANCE.
3. MAGNETS: NEUTRON RADIATION CAUSES CHANGES IN PROPERTIES OF MAGNETS. LITTLE DATA IS AVAILABLE AND NO DATA IS AVAILABLE BELOW ROOM TEMPERATURE.
4. PHENOLICS: UNFILLED PHENOLIC IS SEVERELY DAMAGED IN THE RANGE 10^9 TO 10^{10} ERGS/G (C). LITTLE DAMAGE OCCURS BELOW 5×10^8 ERGS/G (C).
5. PLASTICS: IN GENERAL, PLASTICS ARE MODERATELY TO SEVERELY DAMAGED BY GAMMA FLUENCES ABOVE 10^9 ERGS/G (C). FILLED PHENOLICS ARE MARGINAL AT 10^9 ERGS/G (C).

DESIGN SOLUTIONS TO UNSHIELDED ENVIRONMENT PROBLEMS

1. THERMAL

THE 1000°R TEMPERATURE IN THE MOTOR SHAFT CAN BE REDUCED BY PROVIDING A COOLANT FLOW PATH THROUGH CENTER OF SHAFT. NO OTHER THERMAL PROBLEMS ARE EXPECTED.

2. MATERIALS

- A. THE METAL SPRINGS HAVE SUFFICIENT MARGIN IN THE DESIGN TO FUNCTION ADEQUATELY WITH LOSSES OF UP TO 20 PERCENT IN MOST CASES. DETAILED ANALYSIS WOULD BE REQUIRED TO DETERMINE IF CHANGES SHOULD BE MADE FOR THE BELLEVILLE WASHER APPLICATION.
- B. ELECTRICAL CONDUCTORS ARE NOT CONSIDERED A PROBLEM. CONSTANT CURRENT DEVICES WILL OVERCOME THIS PROBLEM. XE PROBLEMS WERE SIMILAR AND WERE SOLVED.
- C. MAGNETIC MATERIALS WILL NEED A VERY THOROUGH STUDY TO OBTAIN DATA IN ORDER TO PROPERLY IDENTIFY IF A PROBLEM EXISTS. PREVIOUS NERVA EXPERIENCE, SUCH AS WITH THE XE ACTUATOR TORQUEMOTOR, INDICATES NO PROBLEM.
- D. PHENOLICS CAN BE REPLACED BY CERAMICS. THE USE OF FULL COMPLIMENT BEARINGS COULD ALSO SOLVE THIS PROBLEM.
- E. PLASTICS ARE USED IN THE POTENTIOMETER WHICH COULD BE REDESIGNED, SUBSTITUTING CONDUCTIVE PLASTIC WITH A CERMET TYPE OF TECHNOLOGY. THIS WOULD REQUIRE NEW DEVELOPMENT AND A TECHNOLOGY PROGRAM BEYOND XE EXPERIENCE IN THIS AREA.

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ACTUATOR THERMAL ANALYSIS

- PRELIMINARY SCOPING THERMAL ANALYSIS INDICATE ACTUATOR COMPONENT OPERATING TEMPERATURES RANGE FROM 285°R TO 370°R WITH NOMINAL UNSHIELDED HEATING RATES OF 0.33 WATTS/GM-AI AND MAXIMUM TEMPERATURE RANGES FROM 300°R TO 600°R WITH +3 σ UNSHIELDED HEATING RATES OF 0.99 WATTS/GM. CONSERVATIVE HEAT TRANSFER COEFFICIENTS WERE ASSUMED IN THE SCOPING ANALYSIS WITH 250°R H₂ COOLANT.
- AT OPERATING CONDITIONS THE TEMPERATURE OF AVAILABLE COOLANT FOR ACTUATOR IS APPROXIMATELY 250°R.
- AVAILABLE FLOW FOR ACTUATOR COOLING VIA FLOW FROM INNER DRUM COOLANT HOLES THROUGH DRIVE SHAFT TO ACTUATOR IS ESTIMATED TO BE APPROXIMATELY 0.2 LB/SEC PER ACTUATOR.
- AVAILABLE PRESSURE DROP ACROSS THE ACTUATOR FOR THE COOLANT PASSAGE DESIGN CONSIDERATIONS IS ESTIMATED TO BE APPROXIMATELY 20 PSID.

CONCLUSION

- THE ACTUATOR COOLANT PASSAGES CAN BE DESIGNED TO PROVIDE ADEQUATE COOLING OF ALL COMPONENTS WITHIN THE ACTUATOR WITH THE UNSHIELDED OPERATING HEATING RATES.

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ACTUAL OPERATING EXPERIENCE ON ACTUATORS

NRX-A6⁽¹⁾: 1×10^8 RADS; 2×10^{15} N/CM² ($E > 2.9$ MEV); 4×10^{16} N/CM² ($E < 0.4$ EV); NO APPARENT DAMAGE.

PBR⁽²⁾ : 1.3×10^9 RADS; 2×10^{16} N/CM² ($E > 1$ MEV); NO APPARENT DAMAGE.

GTR⁽³⁾ : 1×10^9 RADS; 1.5×10^{16} N/CM² ($E > 2.0$); NO APPARENT DAMAGE.

(1) WANL-TME-1954

(2) NASA-TN-D-5347

(3) WANL-TME-935

EFFECT OF VARIOUS INTERNAL SHIELDS ON THE NOMINAL CALCULATED RADIATION
ENVIRONMENT AT THE CDA FOR 10 HOUR OPERATION AT 1575 MW
(IN-FLIGHT OPERATING ENVIRONMENT; NO ENGINE DISC SHIELD)

(LOCATION: STATION = -53.7 IN., RADIUS = ~23 IN.)

SHIELD	NEUTRON FLUENCE (N/CM ²)			GAMMA DOSE (ERGS/GM-C)	HEATING RATE (W/GM-AI)
	E > 1 MEV	0.4 EV < E < 1 MEV	E < 0.4 EV		
BATH REF. DESIGN ⁽¹⁾ (939J685 REV. E)	1.3×10^{15}	3.2×10^{15}	2.3×10^{13}	1.7×10^9	4.7×10^{-3}
BTiH _{1.5} ⁽²⁾	9.1×10^{13}	1×10^{14}	5.4×10^{12}	5.7×10^8	1.6×10^{-3}
6" BTiH _{1.5} + 3" Pb ⁽³⁾	8.6×10^{15}	4×10^{16}	2.4×10^{15}	8.9×10^8	2.5×10^{-3}
NO SHIELD	3.6×10^{17}	3.6×10^{18}	4×10^{17}	1.12×10^{11}	3.3×10^{-1}

(1) SHIELD ASSEMBLY THICKNESS = 18.25"; WEIGHT OF WALL-TO-WALL SHIELD ASSEMBLY = 3450 LBS.

(2) SHIELD THICKNESS = 18.25"; WEIGHT OF WALL-TO-WALL SHIELD ASSEMBLY = 4500 LBS.

(3) SHIELD WEIGHT = 3400 LBS.

EFFECT OF VARIOUS INTERNAL SHIELDS ON THE NOMINAL CALCULATED
RADIATION ENVIRONMENT AT THE SSCV FOR 10 HOUR OPERATION AT
1575 MW (IN-FLIGHT OPERATING ENVIRONMENT; NO ENGINE DISC SHIELD)

(LOCATION: STATION = -95 IN., RADIUS = ~25 IN.)

SHIELD	NEUTRON FLUENCE (N/CM ²)			GAMMA DOSE (ERGS/GM-C)	HEATING RATE (W/GM-AL)
	E > 1 MEV	0.4 EV < E < 1 MEV	E < 0.4 EV		
BATH REF. DESIGN ⁽¹⁾ (939J685 REV. E)	6×10^{14}	9.4×10^{14}	4.6×10^{13}	1.4×10^9	4.3×10^{-3}
BTiH _{1.5} ⁽²⁾ (15% COOLANT $\sqrt{V}$)	4.2×10^{12}	$4.6 \times 10^{12*}$	$2.5 \times 10^{10*}$	4.7×10^8	1.4×10^{-3}
6" BTiH _{1.5} + 3" Pb ⁽³⁾	4×10^{15}	$1.8 \times 10^{16*}$	$1.1 \times 10^{15*}$	7.3×10^8	2.2×10^{-3}
NO SHIELD	1.5×10^{17}	$1.5 \times 10^{18*}$	1.7×10^{17}	9.1×10^{10}	3.0×10^{-1}

(1) SHIELD ASSEMBLY THICKNESS = 18.25"; WEIGHT OF WALL-TO-WALL SHIELD ASSEMBLY = 3450 LBS.

(2) SHIELD THICKNESS = 18.25"; WEIGHT OF WALL-TO-WALL SHIELD = 4500 LBS.

(3) SHIELD WEIGHT = 3400 LBS.

* RATIOS ASSUMED SAME AS FOR CDA LOCATION

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GAMMA RAY DOSE (REM)

INTEGRATED DOSE (REM) FOR 5 AND 30 MINUTE EXPOSURE TIMES
AT TWO LOCATIONS FOR SEVERAL INTERNAL SHIELD DESIGNS

LOCATION 1 - ABOVE THE SHIELD

TIME AFTER FIRST 10 MIN. BURN	5 MINUTE EXPOSURE *			30 MINUTE EXPOSURE *		
	Unshielded	Ref. Design	B-TiH _{1.5}	Unshielded	Ref. Design	B-TiH _{1.5}
10 DAYS	16.0	0.2	0.07	96.0	1.2	0.42
30 DAYS	4.7	0.068	0.023	28.0	0.41	0.12
TIME AFTER SIX BURNS						
10 DAYS	96.0	1.2	0.42	575.0	7.2	2.5
30 DAYS	28.0	0.41	0.14	170.0	2.5	0.84

LOCATION 7 - PARTIALLY OUTSIDE THE SHADOW OF
THE INTERNAL SHIELD

TIME AFTER FIRST 10 MIN. BURN	5 MINUTE EXPOSURE *			30 MINUTE EXPOSURE *		
	Unshielded	Ref. Design	B-TiH _{1.5}	Unshielded	Ref. Design	B-TiH _{1.5}
10 DAYS	30.0	11.0	10.0	180.0	66.0	60.0
30 DAYS	8.7	3.3	3.0	52.0	20.0	18.0
TIME AFTER SIX BURNS						
10 DAYS	180.0	66.0	60.0	1080.0	390.0	360.0
30 DAYS	52.0	20.0	18.0	312.0	120.0	108.0

* IT WAS ESTIMATED THAT AN ACTUATOR COULD BE REMOVED IN 5 MINUTES IN SPACE AND A NEW ACTUATOR INSTALLED IN 30 MINUTES.

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TYPES OF FUEL STUDIES

- 2.1 INTERMEDIATE CTE FUEL ELEMENT
A. 19, 0.102-INCH DIAMETER FLOW CHANNELS AND A 0.753-INCH HEX.
B. PYROCARBON COATED UC₂ FUEL BEADS DISPERSED THROUGH A GRAPHITE MATRIX OF INTERMEDIATE CTE.
- 2.2 HIGH CTE FUEL ELEMENT
SIMILAR TO 2.1, BUT EMPLOYS A GRAPHITE MATRIX OF HIGH CTE.
- 2.3 HYBRID FUEL ELEMENT
CONSISTS OF A 35-INCH LONG FORWARD SECTION OF FUELED GRAPHITE (DEFINED BY 2.2) BRAZED TO A 17-INCH LONG SECTION OF SOLID SOLUTION UC-ZrC-C COMPOSITE.
- 2.4 MODIFIED HYBRID FUEL ELEMENT
SIMILAR TO 2.3, BUT HAS 37, 0.077-INCH DIAMETER FLOW CHANNELS AND A 0.799-INCH HEX.
- 2.5 ALL COMPOSITE FUEL ELEMENT
SIMILAR TO 2.1, BUT COMPOSED ENTIRELY OF UC-ZrC-C COMPOSITE.
- 2.6 ALL CARBIDE FUEL ELEMENT
SIMILAR TO 2.1, BUT COMPOSED ENTIRELY OF UC-ZrC.

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~~RESTRICTED DATA~~
Atomic Energy Act - 1954

ESTIMATED BATH SHIELD WEIGHTS AND DELTA SHIELDING WEIGHTS WHICH
GIVE A CONSTANT DOSE RATE⁽¹⁾ AT 30 FEET FROM THE
CORE CENTER ON THE REACTOR CENTERLINE

CORE DIAMETER = 37.4 INCHES
CORE LENGTH = 52.0 INCHES
VOID FRACTION = 32 V/O

<u>FUEL ELEMENT CONFIGURATION</u>	<u>SHIELD WEIGHT (LBS)</u>	<u>DELTA SHIELD WEIGHT (LBS)</u>
2.1 INTERMEDIATE CTE FUEL ELEMENT	3850	---
2.2 HIGH CTE FUEL ELEMENT	3897	+ 47
2.3 HYBRID FUEL ELEMENT	3897	+ 47
2.4 MODIFIED HYBRID FUEL ELEMENT	3897	+ 47
2.5 ALL COMPOSITE FUEL ELEMENT	3638	-212
2.6 ALL CARBIDE FUEL ELEMENT	3431	-419

(1) THE CONSTANT DOSE RATE IS THE MEDIUM EXPANSION DESIGN LEVEL FOR EITHER GAMMA
RAYS OR NEUTRONS

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~~RESTRICTED DATA~~
Atomic Energy Act - 1954

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ESTIMATED GAMMA RAY AND FAST NEUTRON DOSE RATE
AT 30 FEET FROM THE CORE CENTER ON THE REACTOR CENTERLINE

CORE DIAMETER = 37.4 INCHES
CORE LENGTH = 52.0 INCHES
VOID FRACTION = 32 V/O

<u>FUEL ELEMENT CONFIGURATION</u>	<u>ESTIMATED DOSE RATE (REM/HR)⁽¹⁾</u>	
	<u>GAMMA</u>	<u>FAST NEUTRON</u>
2.1 INTERMEDIATE CTE FUEL ELEMENT	1.1 (5)	5.5 (5)
2.2 HIGH CTE FUEL ELEMENT	1.1 (5)	5.7 (5)
2.3 HYBRID FUEL ELEMENT	1.1 (5)	5.7 (5)
2.4 MODIFIED HYBRID FUEL ELEMENT	1.1 (5)	5.7 (5)
2.5 ALL COMPOSITE FUEL ELEMENT	7.9 (4)	4.4 (5)
2.6 ALL CARBIDE FUEL ELEMENT	5.2 (4)	3.5 (5)

(1) NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

ESTIMATED BATH SHIELD WEIGHTS AND DELTA SHIELDING WEIGHTS WHICH
GIVE A CONSTANT DOSE RATE⁽¹⁾ AT 30 FEET FROM THE
CORE CENTER ON THE REACTOR CENTERLINE

CORE DIAMETER = 51.0 INCHES
CORE LENGTH = 52.0 INCHES
VOID FRACTION = 32 V/O

<u>FUEL ELEMENT CONFIGURATION</u>	<u>SHIELD WEIGHT (LBS)</u>	<u>DELTA SHIELD WEIGHT (LBS)</u>
2.1 INTERMEDIATE CTE FUEL ELEMENT	6584	---
2.2 HIGH CTE FUEL ELEMENT	6664	+ 80
2.3 HYBRID FUEL ELEMENT	6664	+ 80
2.4 MODIFIED HYBRID FUEL ELEMENT	6664	+ 80
2.5 ALL COMPOSITE FUEL ELEMENT	6221	-363
2.6 ALL CARBIDE FUEL ELEMENT	5867	-717

(1) THE CONSTANT DOSE RATE IS THE MEDIUM EXPANSION DESIGN LEVEL FOR EITHER GAMMA
RAYS OR NEUTRONS.

ESTIMATED GAMMA RAY AND FAST NEUTRON DOSE RATE AT 30 FEET
FROM THE CORE CENTER ON THE REACTOR CENTERLINE

CORE DIAMETER = 51.0 INCHES
CORE LENGTH = 52.0 INCHES
VOID FRACTION = 32 V/O

<u>FUEL ELEMENT CONFIGURATION</u>	<u>ESTIMATED DOSE RATE (REM/HR)⁽¹⁾</u>	
	<u>GAMMA</u>	<u>FAST NEUTRON</u>
2.1 INTERMEDIATE CTE FUEL ELEMENT	1.2 (5)	5.2 (5)
2.2 HIGH CTE FUEL ELEMENT	1.2 (5)	5.4 (5)
2.3 HYBRID FUEL ELEMENT	1.2 (5)	5.4 (5)
2.4 MODIFIED HYBRID FUEL ELEMENT	1.2 (5)	5.4 (5)
2.5 ALL COMPOSITE FUEL ELEMENT	8.2 (4)	4.2 (5)
2.6 ALL CARBIDE FUEL ELEMENT	5.7 (4)	3.3 (5)

(1) NUMBERS IN PARENTHESES REFER TO POWERS OF TEN