JUNE 6 1983

DMB 016

Dockets Nos. 50-269, 50-270
and 50-287

LICENSEE: Duke Power Company (DPC)

FACILITY: Oconee Nuclear Station, Units Nos. 1, 2 and 3

SUBJECT: SUMMARY OF MEETING HELD ON MAY 12, 1983 WITH REPRESENTATIVES OF DPC AND BABCOCK & WILCOX TO DISCUSS NEUTRON DOSIMETRY AND UPPERSHELF MATERIAL PROPERTIES

The meeting was held to clarify the additional information requested by NRC letter dated March 22, 1983 in regard to the NRC review of the Oconee neutron dosimetry provided in BAW-1697 and BAW-1699. The attendees list and a copy of the meeting agenda utilized is enclosed.

Discussion

Bob Gill (DPC) announced at the beginning of the meeting his tentative schedule to submit the written responses rquested by NRC in the March 22, 1983 letter by May 23, 1983. [Project Manager's Note: This submittal date will probably be delayed one week.] In addressing the four areas of concern to NRC, the presentation made by Whitmarsh and Lowe (B&W) presented the case that the analysis made to date took into sufficient account the plant specific similarities for neutron fluence and that due to their method of measurement the rotation of the capsule does not affect the fluence values obtained. Although B&W say they can support their uncertainty values used in the Oconee fluence analysis, NRC questioned the precision of the individual elements that comprise the overall cumulative uncertainty values. NRC inquired of B&W on the advisability of updating the predictions used in uper shelf life. Updating is being considered by B&W for future work. NRC still questions the predictive value of B&W's model in addressing weld metals.

Original signed by

John F. Suermann, Project Manager
Operating Reactors Branch #4, DL

Enclosures:
As Stated

cc w/enclosures:
See next page

8306200185 830606
PDR ADOCK 05000269
P PDR

OFFICE	ORB#4:DL						
SURNAME	JFSuermann;cf						
DATE	6/3/83						

JFS

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

MEETING SUMMARY DISTRIBUTION

Licensee:

*Copies also sent to those people on service (cc) list for subject plant(s).

Docket File
NRC PDR
L PDR
ORB#4 Rdg
Project Manager-JSuermann
JStolz
BGrimes (Emerg. Preparedness only)

OELD
NSIC
ELJordan, IE
JMTaylor, IE
ACRS (10)

NRC Meeting Participants:
LLois
BElliot
PRandall

MEETING OF MAY 12, 1983
OCONEE NEUTRON DOSIMETRY

NRC	DPC	B&W
JSuermann	RGill	CWhitmarsh
LLois	JPetty	ALowe, Jr.
BElliot		CHudson
PRandall		HBehnke
		CChagnon

Agenda

Duke Power/NRC/B&W

10:00 a.m. May 12, 1983 Room P110 Phillips Building

Bethesda, Maryland

Introduction	B.J. Elliot - NRC R.L. Gill - DPCo
Objective of B&W Presentation	C.J. Hudson - B&W
Review of B&W Fluence Analysis Calculational Procedure	C.L. Whitmarsh - B&W
Identify Sources of Uncertainty	C.L. Whitmarsh
Status of Fluence Analysis at B&W	C.L. Whitmarsh
NRC Concerns for Oconee Fluence Plant Similarity Capsule Rotation Uncertainty Values Use Limit	C.L. Whitmarsh/A.L. Lowe
Future Action NRC Procedures Utility/Vendor Effort	C.L. Whitmarsh/A.L. Lowe
Summary	R.L. Gill - DPCo

DPCO/NRC/B&W MEETING OBJECTIVE

- REVIEW FLUENCE ANALYSIS COMPLETED FOR OCONEE

- ANSWER THE QUESTIONS ASKED OF DUKE POWER BY 3/22/83, LETTER FROM J. F. STOLZ TO H. B. TUCKER, DOCKET NOS. 50-269, 50-270 AND 50-287.

- OBTAIN NRC REGULATORY POSITION AND PLANS FOR REACTOR VESSEL FLUENCE DETERMINATION

CALCULATIONAL MODEL INPUTS

GEOMETRY

MATERIALS

POWER DISTRIBUTION

TRANSPORT CODE PARAMETER

DOSIMETER ACTIVITY CALCULATION

DOSIMETER ACTIVITY MEASUREMENT

REACTOR VESSEL ARRANGEMENT

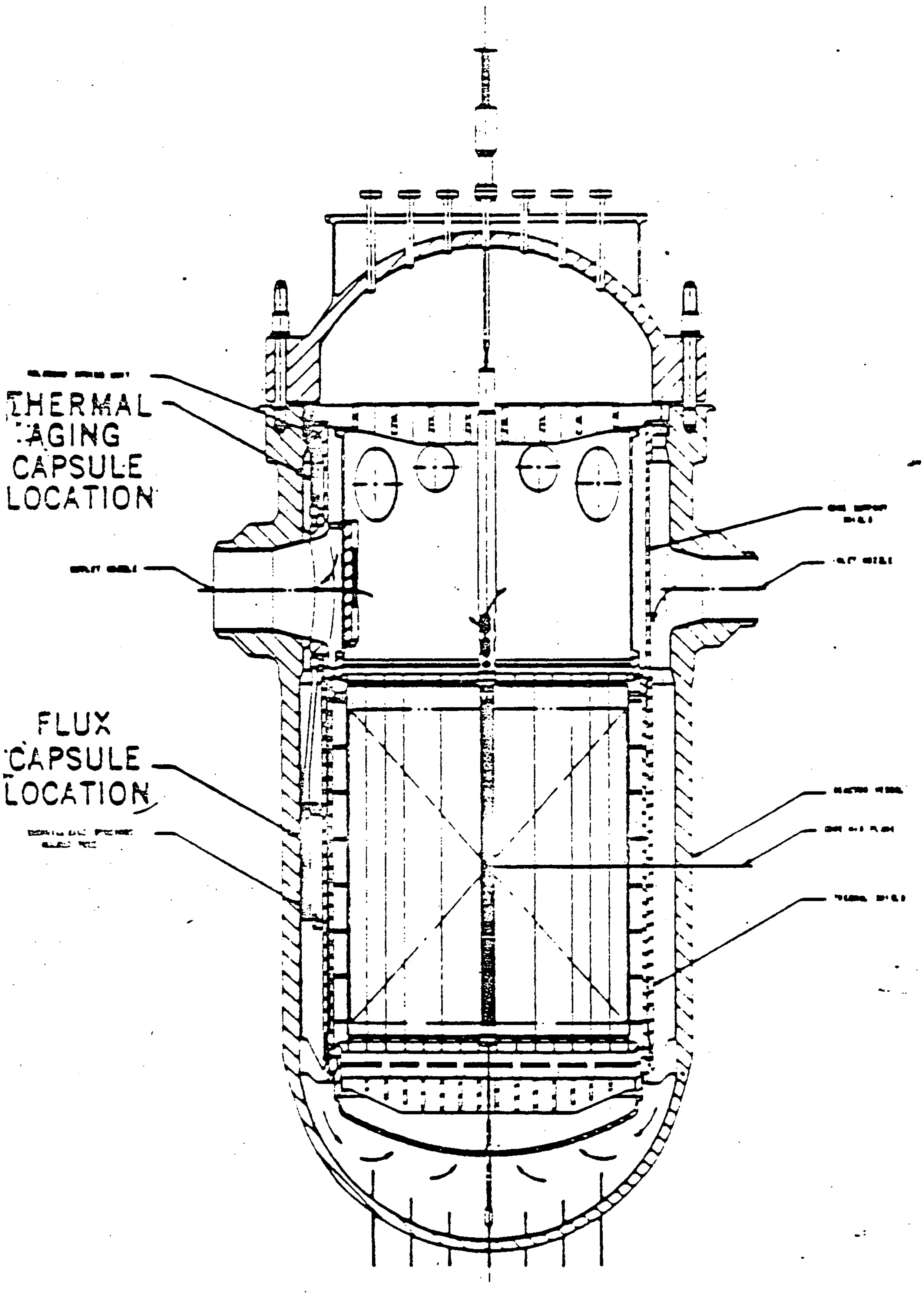


Figure 3-.. Reactor Vessel Cross Section Showing Location of Crystal River Unit 3, Capsule CR3-B

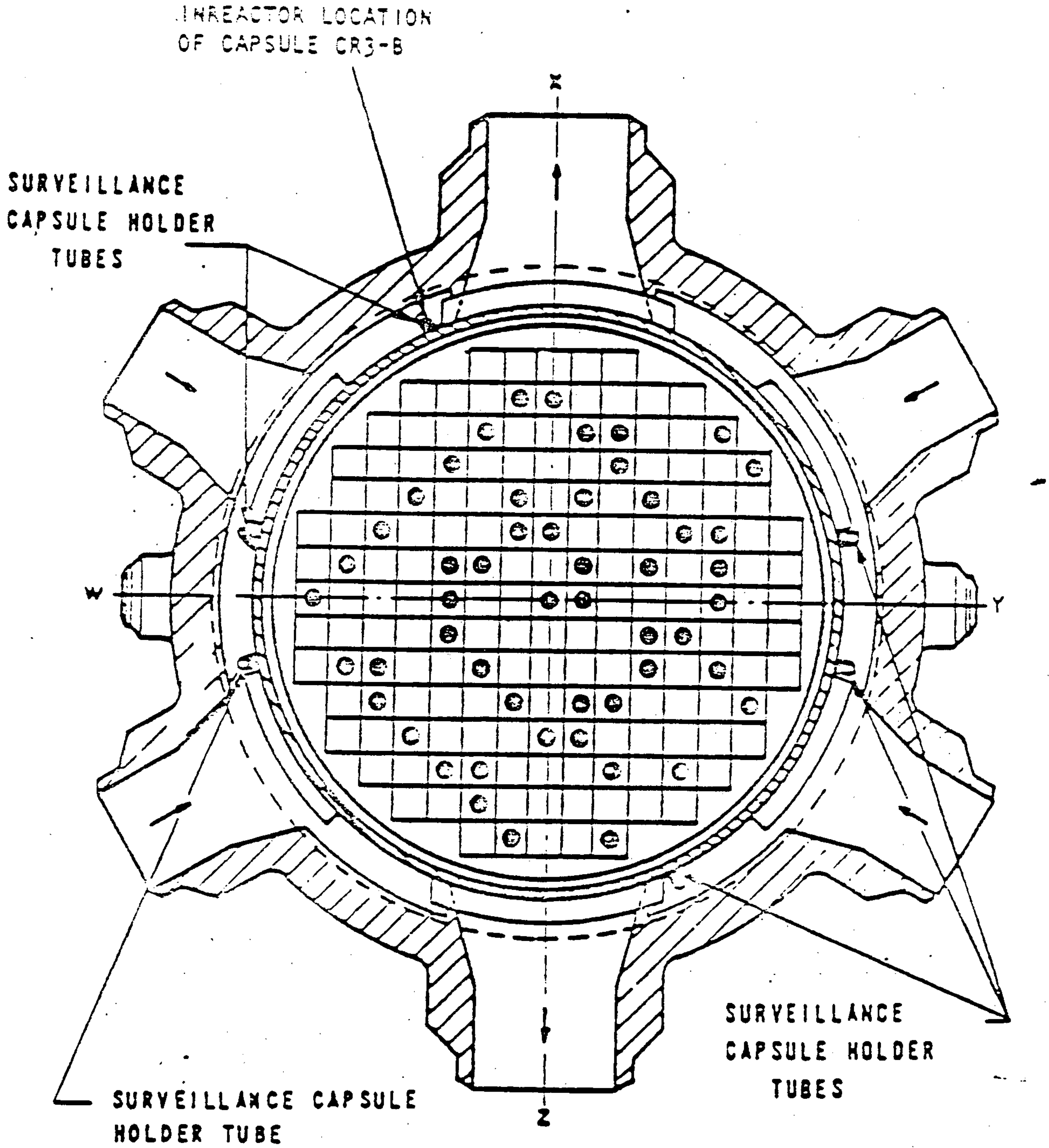
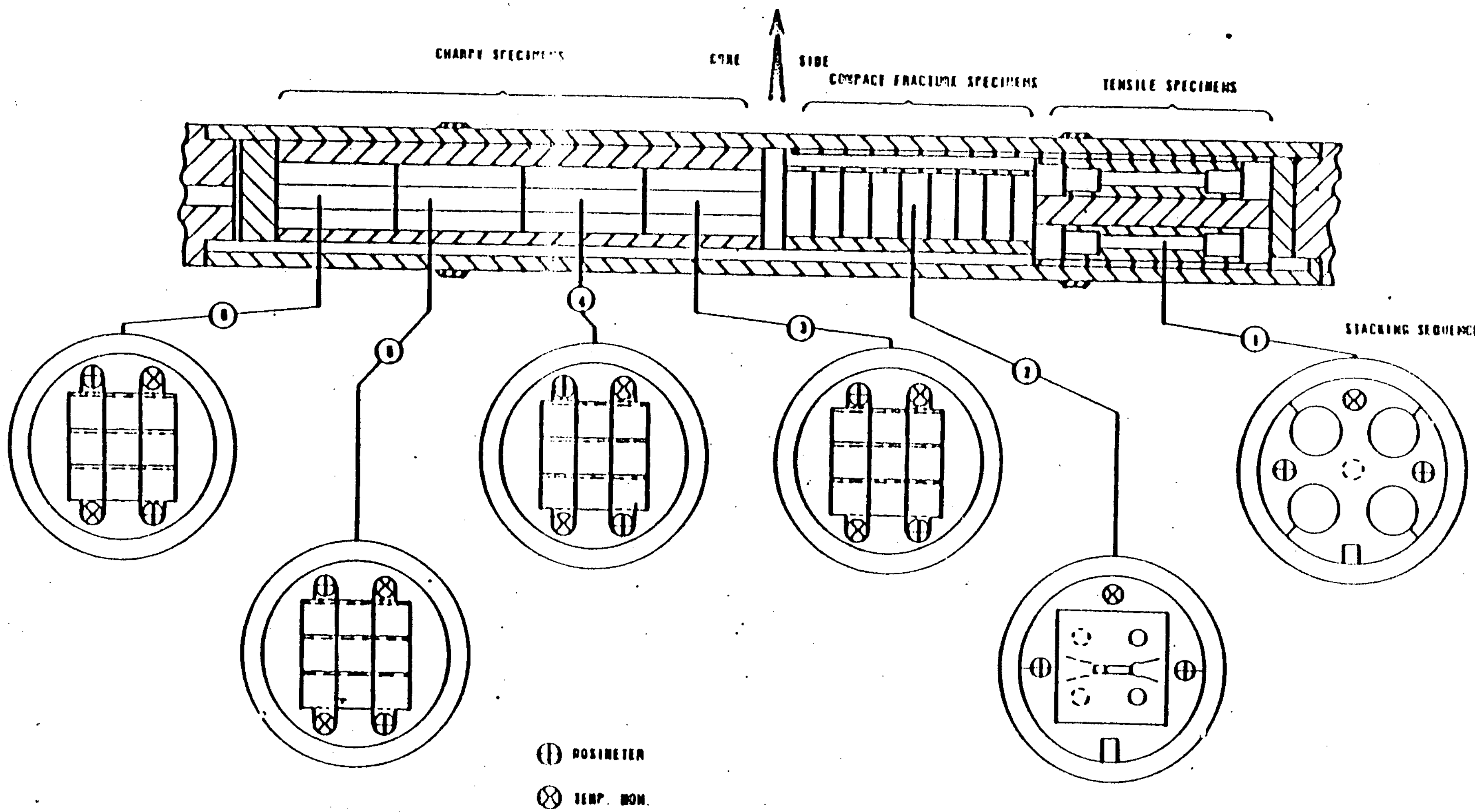
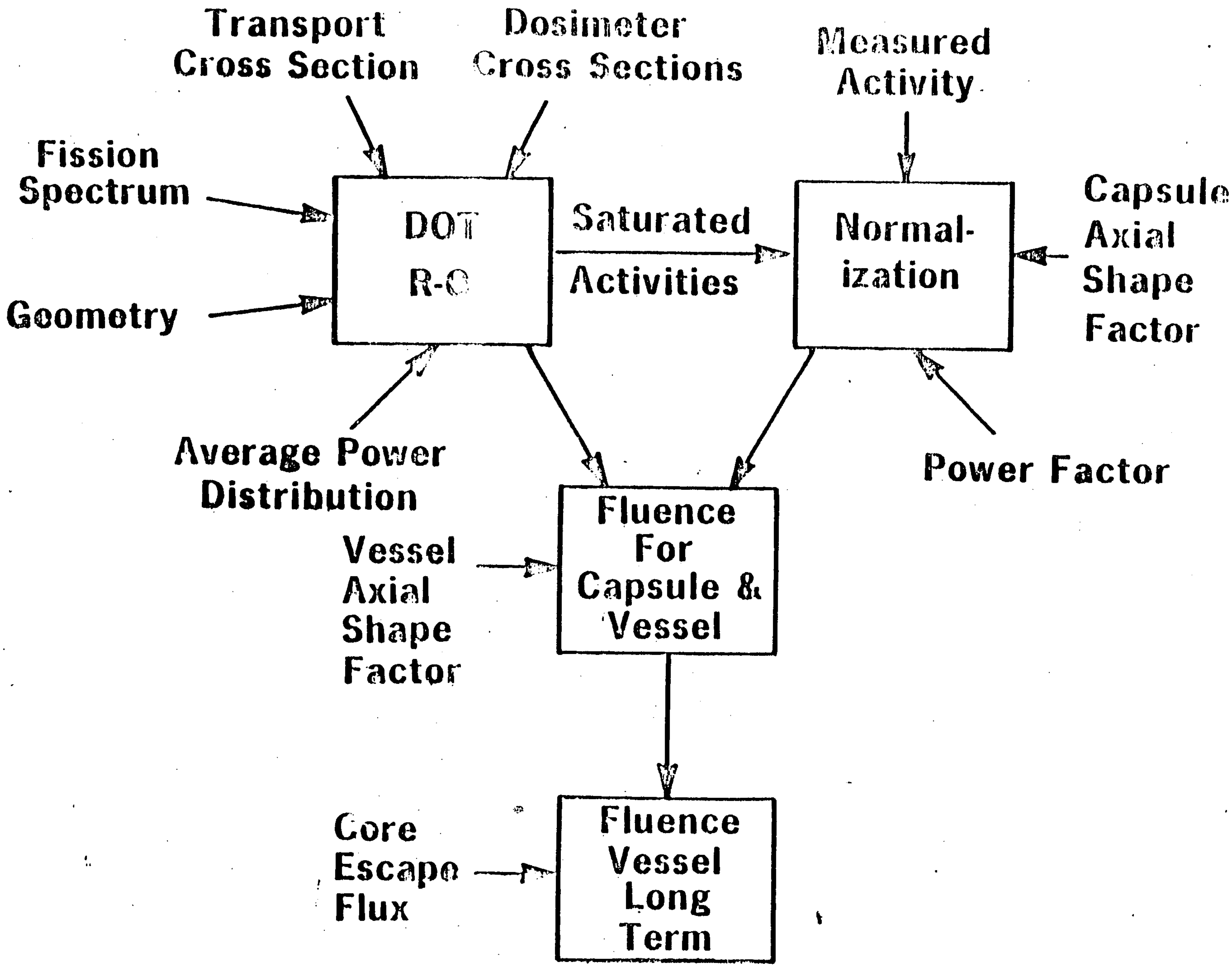
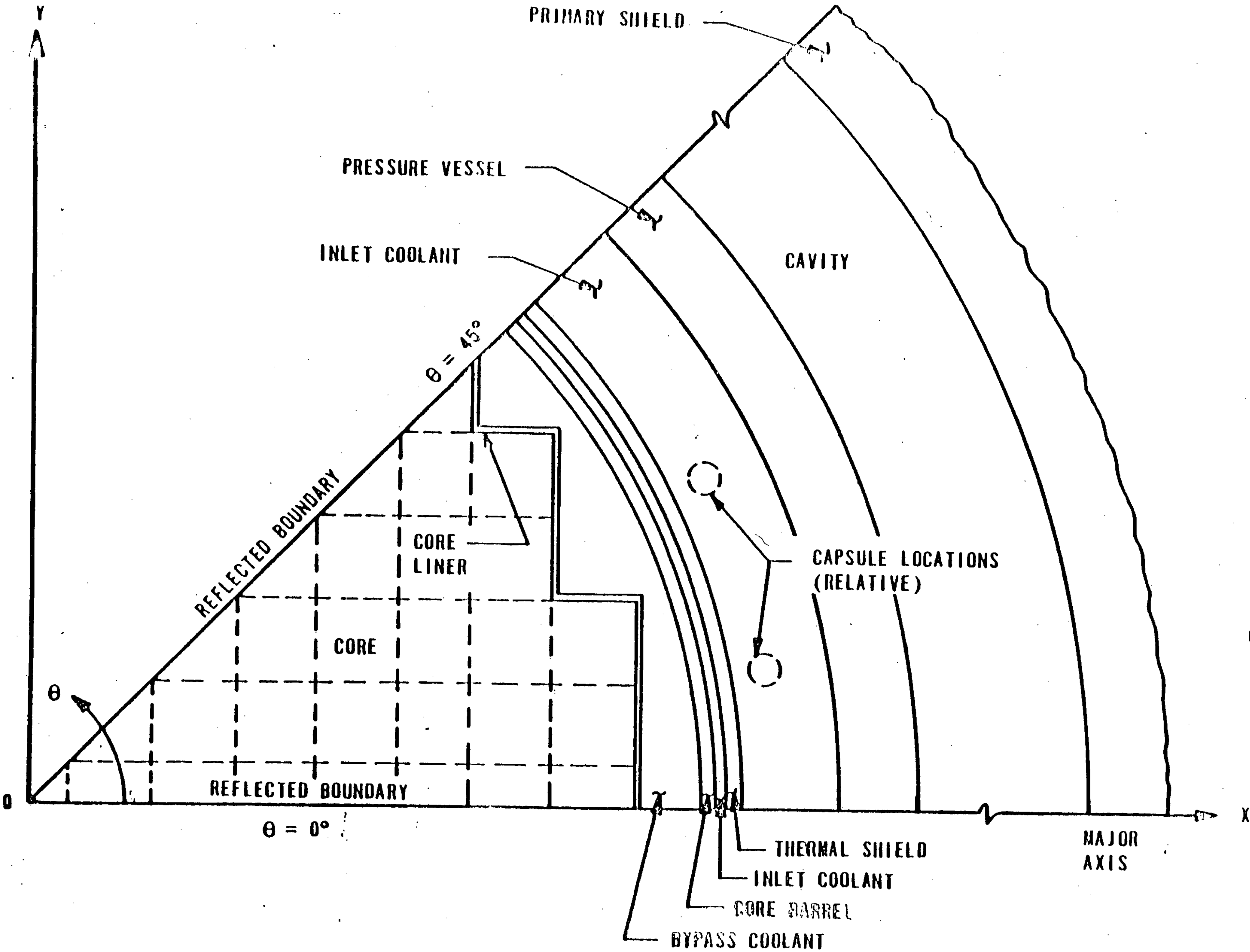


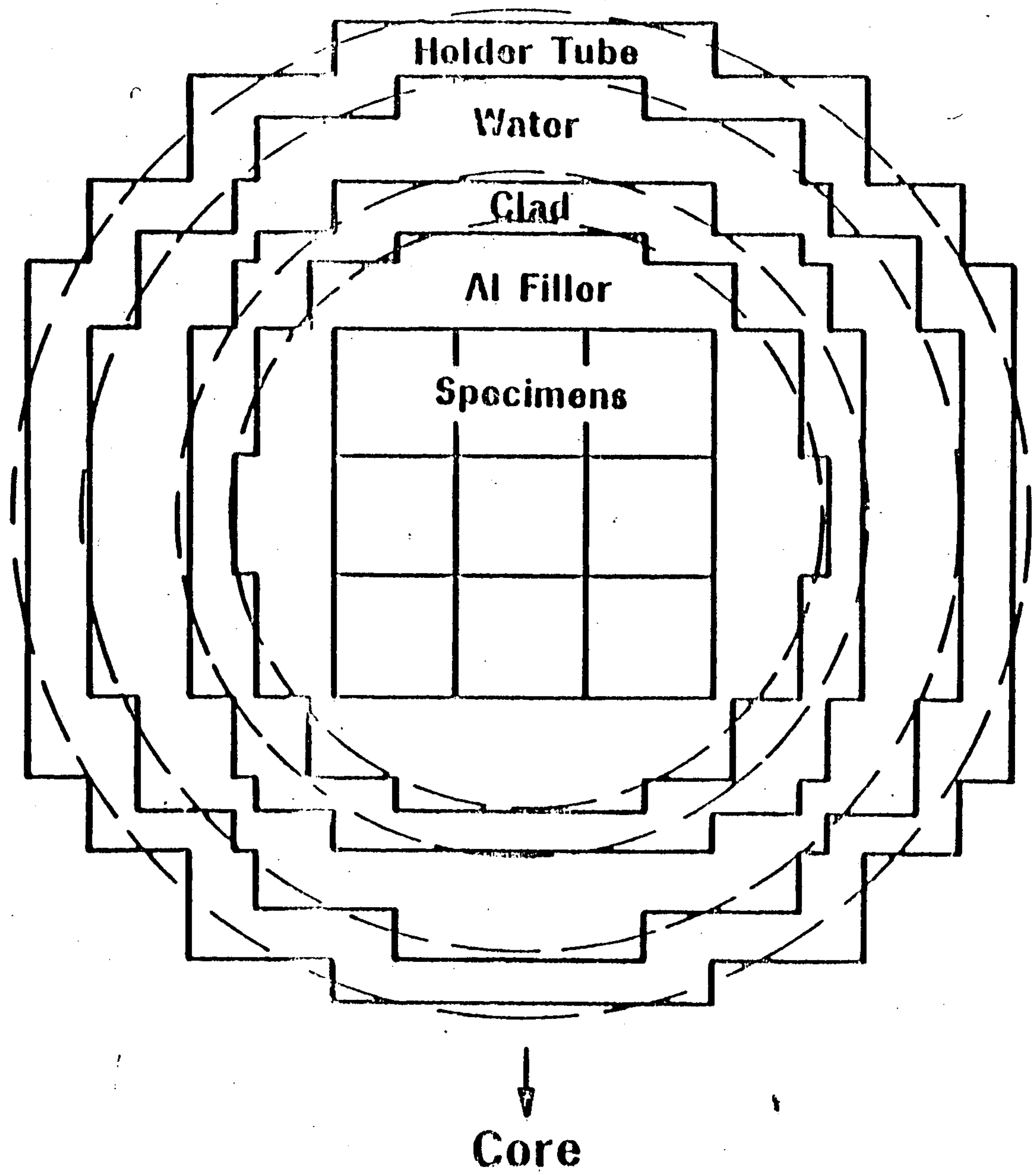
Figure 3-2. Surveillance Capsule Arrangement — With Compact Tension Specimens




Transport Cross Section
Dosimeter Cross Sections
Measured Activity
Fission Spectrum
DOT R-C
Saturated Activities
Normal-ization
Capsule Axial Shape Factor
Geometry
Average Power Distribution
Power Factor
Fluence For Capsule & Vessel
Vessel Axial Shape Factor
Core Escape Flux
Fluence Vessel Long Term



Y
PRIMARY SHIELD
PRESSURE VESSEL
INLET COOLANT
CAVITY
θ = 45°
REFLECTED BOUNDARY
CORE LINER
CAPSULE LOCATIONS (RELATIVE)
CORE
θ
0
REFLECTED BOUNDARY
θ = 0°
X
THERMAL SHIELD
INLET COOLANT
CORE BARREL
BYPASS COOLANT
MAJOR AXIS


Holder Tube
Water
Clad
Al Filler
Specimens
Core

CALCULATIONAL MODEL INPUTS

GEOMETRY

MATERIALS

POWER DISTRIBUTION

TRANSPORT CODE PARAMETER

DOSIMETER ACTIVITY CALCULATION

DOSIMETER ACTIVITY MEASUREMENT

GEOMETRY INPUTS

CAPSULE LOCATION

- Radia~~tion~~lly $\sim$15%/cm
- Azimuthal $\sim$1%/deg (4 cm)

SIMULATION OF NON-CYLINDRICAL BOUNDARIES

- Streaming
- Attenuation

EFFECT OF TEMPERATURE

- Operating condition

DIMENSIONAL TOLERANCES

- Nominal dimensions

MATERIALS INPUTS

METAL COMPOSITION

Nominal values

CORE COMPOSITION

Mid-cycle values

COOLANT TEMPERATURE IN BYPASS REGION

Between inlet and core avg. temp.

6%/24°F

INLET COOLANT TEMPERATURE

.2%/°F at capsule

.4%/°F at vessel inner surface

Function of reactor power

POWER DISTRIBUTION INPUTS

SENSITIVITY NEAR CORE EDGE

Linear proportionality

Difficult to calculate

BOC TO EOC

~15% variation

Avg. static value

CYCLE TO CYCLE

~20% in early cycles

Dependent on fuel management

FISSION SPECTRUM

Fresh fuel primarily ^{235}U

Multiburned fuel 50-70% ^{239}Pu

TRANSPORT CODE PARAMETERS

P_3 Cross Sections

S_8 Quadrature

Mesh Spacing

- Smaller is more accurate
- Smaller is more costly

Microscopic Cross Sections

- Fundamental point data
- Broad group averaging
- Energy groups

Macroscopic Cross Sections

- Material density

DOSIMETER ACTIVITY CALCULATION

$$A = K \sum \sigma(E)\phi(E) \sum F\,(1-e^{-\lambda t_1})e^{-\lambda t_2}$$

NON SATURATION

Reactor power vs. time

Isotope half life

AXIAL FLUX SHAPE

RPD in nearby fuel assembly

SHORT LIVED ISOTOPES

Insensitive to flux early in irradiation

ENERGY RESPONSE

Only fission dosimeters responsive in
$2.5 \geq E > 1.0$ MeV

>60% of fast flux in capsule in

$2.5 \geq E > 1.0$ MeV

DOSIMETER REACTION CROSS SECTIONS

ENDF/BV Data

Broad group averaging

FISSION YIELDS

ENDF/BV Data

DOSIMETER ACTIVITY MEASUREMENT

COUNTING

- Statistics
- Efficiency

IMPURITY REACTIONS

- Competing absorption
- γ, f in fission dosimeters

MATERIAL COMPOSITION

- Weight % of target isotope

SOURCES OF UNCERTAINTY TO VESSEL FLUENCE
EFFECT OF CAPSULE NORMALIZATION

ELIMINATED OR REDUCED SIGNIFICANTLY

- Core boundary simulation
- Temperature effect on dimensions
- Coolant temperature in bypass region
- Core and SS304 compositions
- Power
- Axial flux distribution

UNAFFECTED OR PARTIALLY REDUCED

Capsule location		
Inlet coolant temperature		
Power distribution		
Fission spectrum		
Microscopic cross sections	(PCA)	(P)
Macroscopic cross sections	(PCA)	(P)
Dimensional tolerances		
Mesh spacing	(PCA)	(P)
Order of scattering	(PCA)	(P)
Quadrature	(PCA)	(P)

ADDED EFFECTS

Activity measurement		(P)
Activity to fluence conversion	(PCA)	(P)
Flux perturbation		(P)

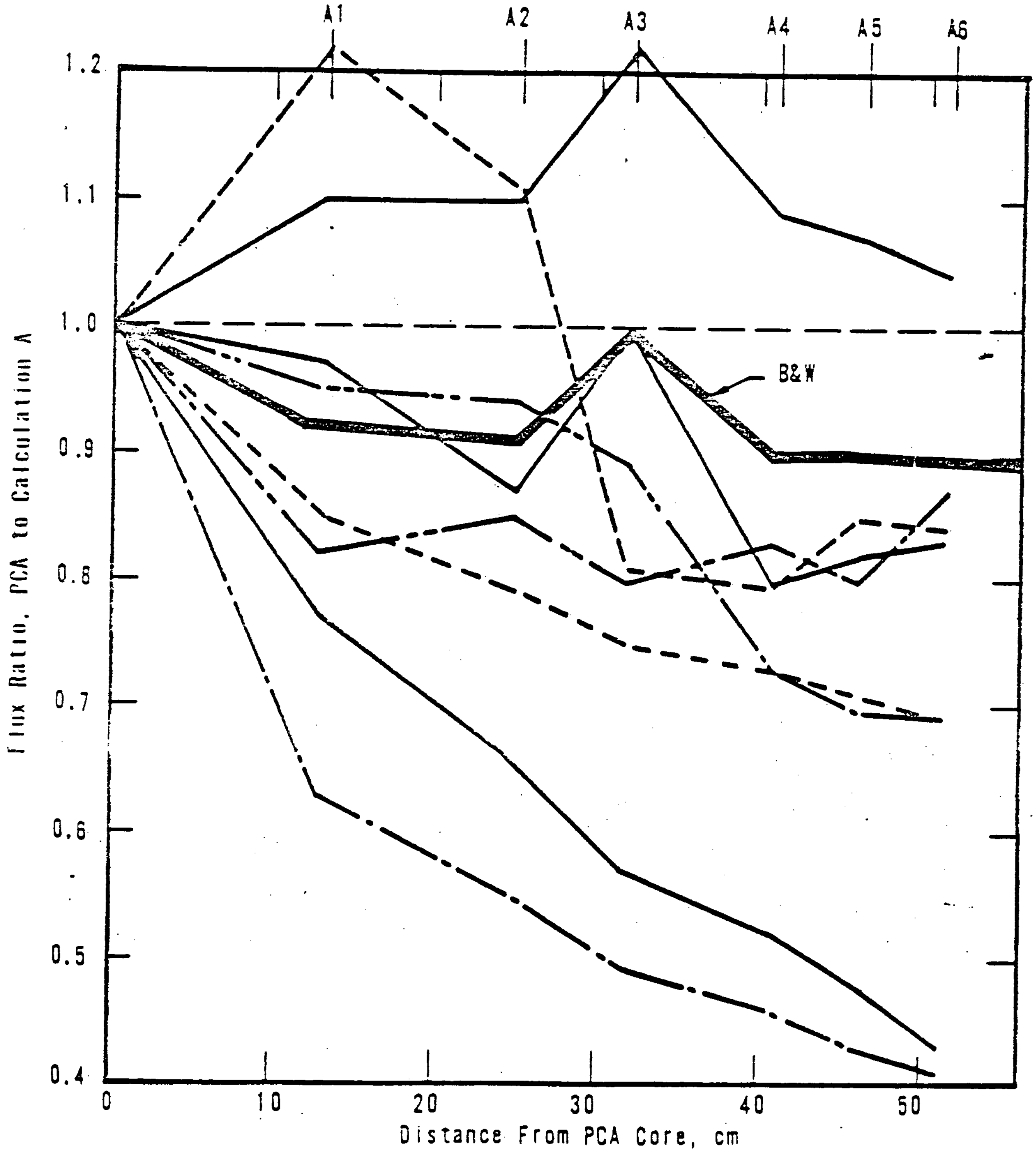

A1
A2
A3
A4
A5
A6
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
B&W
Flux Ratio, PCA to Calculation A
0
10
20
30
40
50
Distance From PCA Core, cm

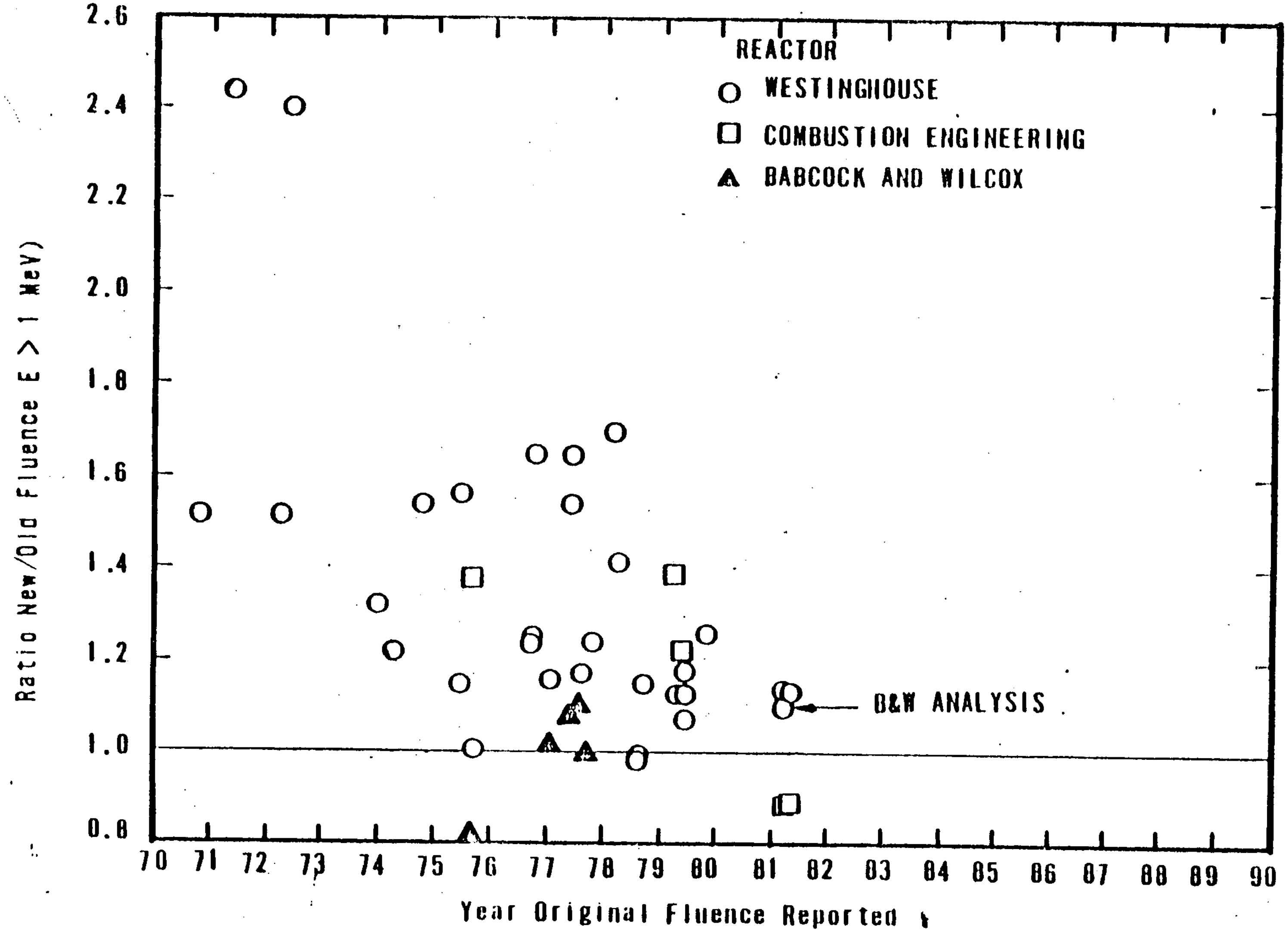
TABLE 3.4

REEVALUATED EXPOSURE VALUES AND THEIR UNCERTAINTY FOR LWR PRESSURE VESSEL SURVEILLANCE CAPSULES

(Revision of Reference 40 data)

Plant	Unit	Capsule	Fluence (at > 1 MeV) (n/cm²) Old	New [% (1σ)]	New/Old	dpa [% (1σ)]	dpa/φt New	dpa/s	Exposure* Time (s)
Westinghouse									
Conn. Yankee		A	2.08 + 18	3.17 + 18 (12)**	1.52	4.89-03 (12)	1.54-21	9.18-11	5.233 + 07
Conn. Yankee		F	4.04 + 18	6.17 + 18 (24)	1.53	9.70-03 (27)	1.57-21	1.27-10	7.651 + 07
Conn. Yankee		H	1.79 + 19	2.06 + 19 (25)	1.15	3.38-02 (28)	1.64-21	1.42-10	2.380 + 08
San Onofre		A	1.20 + 19	2.93 + 19 (22)	2.44	5.04-02 (27)	1.72-21	8.66-10	5.824 + 07
San Onofre		B	2.36 + 19	5.66 + 19 (26)	2.40	9.51-02 (29)	1.68-21	1.07-09	8.881 + 07
San Onofre		F	5.14 + 19	5.81 + 19 (14)	1.13	9.79-02 (21)	1.69-21	4.02-10	2.438 + 08
Turkey Pt.	3	S	1.41 + 19	1.66 + 19 (25)	1.18	2.66-02 (27)	1.60-21	2.42-10	1.096 + 08
Turkey Pt.	3	T	5.68 + 18	7.05 + 18 (10)	1.24	1.09-02 (12)	1.55-21	4.76-10	2.222 + 07
Turkey Pt.	4	S	1.25 + 19	1.34 + 19 (25)	1.07	2.22-02 (27)	1.66-21	2.66-10	1.621 + 08
Turkey Pt.	4	T	6.06 + 18	7.58 + 18 (13)	1.25	1.32-02 (13)	1.74-21	3.53-10	3.728 + 07
H. B. Robinson	2	S	3.02 + 18	3.99 + 18 (24)	1.32	6.99-03 (27)	1.75-21	1.66-10	4.209 + 07
H. B. Robinson	2	V	4.51 + 18	7.43 + 18 (22)	1.65	1.19-02 (25)	1.60-21	1.14-10	1.060 + 08
Surry	1	T	2.50 + 18	2.88 + 18 (9)	1.15	4.56-03 (12)	1.58-21	1.35-10	3.378 + 07
Surry	2	X	3.02 + 18	3.05 + 18 (11)	1.01	4.81-03 (13)	1.58-21	1.30-10	3.687 + 07
North Anna	1	V	2.49 + 18	2.74 + 18 (9)	1.10	4.17-03 (11)	1.52-21	1.17-10	3.570 + 07
Pr. Island	1	V	5.21 + 18	6.09 + 18 (11)	1.17	1.06-02 (16)	1.72-21	2.46-10	4.248 + 07
Pr. Island	2	V	5.00 + 18	6.80 + 18 (10)	1.24	1.19-02 (13)	1.75-21	2.71-10	4.394 + 07
R. E. Ginna	1	R	7.60 + 18	1.17 + 19 (10)	1.54	2.18-02 (14)	1.86-21	2.62-10	8.328 + 07
R. E. Ginna	1	V	4.90 + 18	5.98 + 18 (14)	1.22	1.02-02 (22)	1.71-21	2.22-10	4.612 + 07
Kewaunee		V	5.59 + 18	6.46 + 18 (10)	1.16	1.16-02 (13)	1.80-21	2.86-10	4.067 + 07
Pt. Beach	1	S	—	8.51 + 18 (10)	—	1.48-02 (13)	1.74-21	1.27-10	1.163 + 08
Pt. Beach	1	R	2.22 + 19	2.17 + 19 (10)	0.98	4.41-02 (14)	2.03-21	2.70-10	1.632 + 08
Pt. Beach	2	T	9.45 + 18	9.47 + 18 (10)	1.00	1.59-02 (13)	1.68-21	1.46-10	1.087 + 08
Pt. Beach	2	V	4.74 + 18	7.33 + 18 (11)	1.56	1.23-02 (13)	1.68-21	2.56-10	4.805 + 07
Pt. Beach	2	R	2.01 + 19	2.54 + 19 (10)	1.26	4.68-02 (14)	1.84-21	2.85-10	1.640 + 08
D. C. Cook	1	T	1.80 + 18	2.78 + 18 (22)	1.54	4.61-03 (26)	1.66-21	1.16-10	3.991 + 07
Indian Pt.	2	T	2.02 + 18	3.34 + 18 (22)	1.65	5.49-03 (27)	1.64-21	1.23-10	4.473 + 07
Indian Pt.	3	T	2.92 + 18	3.30 + 18 (22)	1.13	5.38-03 (26)	1.63-21	1.28-10	4.211 + 07
Zion	1	T	1.80 + 18	3.06 + 18 (10)	1.70	4.97-03 (12)	1.62-21	1.31-10	3.789 + 07
Zion	1	U	8.92 + 18	1.02 + 19 (10)	1.14	1.68-02 (13)	1.65-21	1.49-10	1.123 + 08
Zion	2	U	2.00 + 18	2.82 + 18 (9)	1.41	4.54-03 (12)	1.61-21	1.13-10	4.007 + 07
Salem	1	T	2.56 + 18	2.91 + 18 (22)	1.14	4.77-03 (26)	1.64-21	1.39-10	3.426 + 07
Combustion Engineering									
Palisades		A240	4.40 + 19	6.10 + 19 (23)	1.39	9.77-02 (28)	1.60-21	1.37-09	7.130 + 07
Fort Calhoun		W225	5.10 + 18	6.22 + 18 (15)	1.22	9.20-03 (18)	1.48-21	1.12-10	8.191 + 07
Maine Yankee		1	1.30 + 19	1.79 + 19 (19)	1.38	2.43-02 (23)	1.64-21	1.05-09	2.777 + 07
Maine Yankee		2	8.84 + 19	7.85 + 19 (13)	0.89	1.25-01 (18)	1.59-21	8.61-10	1.446 + 08
Maine Yankee		W263	6.90 + 18	6.12 + 18 (13)	0.89	9.21-03 (15)	1.50-21	6.37-11	1.446 + 08
Babcock & Wilcox									
Oconee	1	F	8.70 + 17	7.10 + 17 (21)	0.82	9.83-04 (20)	1.38-21	3.74-11	2.629 + 07
Oconee	1	E	1.50 + 18	1.50 + 18 (10)	1.00	2.11-03 (10)	1.41-21	4.07-11	5.186 + 07
Oconee	2	C	9.43 + 17	1.02 + 18 (10)	1.08	1.50-03 (11)	1.47-21	3.95-11	3.802 + 07
Oconee	3	A	7.39 + 17	8.10 + 17 (10)	1.10	1.15-03 (11)	1.42-21	3.86-11	2.983 + 07
Three Mile Is.	1	E	1.07 + 18	1.09 + 18 (9)	1.02	1.53-03 (9)	1.40-21	3.80-11	4.026 + 07
				avg	1.25				

*Equivalent constant power level exposure time.
**3.17 + 18 means 3.17×10^{18} with a 12% (1σ) uncertainty.


REACTOR
WESTINGHOUSE
COMBUSTION ENGINEERING
BABCOCK AND WILCOX
B&W ANALYSIS
Ratio New/Old Fluence E > 1 MeV)
Year Original Fluence Reported
2.6
2.4
2.2
2.0
1.8
1.6
1.4
1.2
1.0
0.8
70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90

PROGRAM TO IMPROVE FLUENCE ANALYSIS - B&W OWNERS GROUP

TO DATE

- SSC-1 Test
- PCA Blind Test
- ASTM E10.05

IN PROGRESS

- Capsule Perturbation Test
- DPA

PLANNED

- PSF Blind Test
- ENDF/BIV or BV Transport Cross Sections
- Cavity Dosimetry
- In-Reactor Benchmark Test

REACTOR PROPERTIES

PERTINENT TO VESSEL FLUENCE

1. CORE SHAPE

2. FUEL

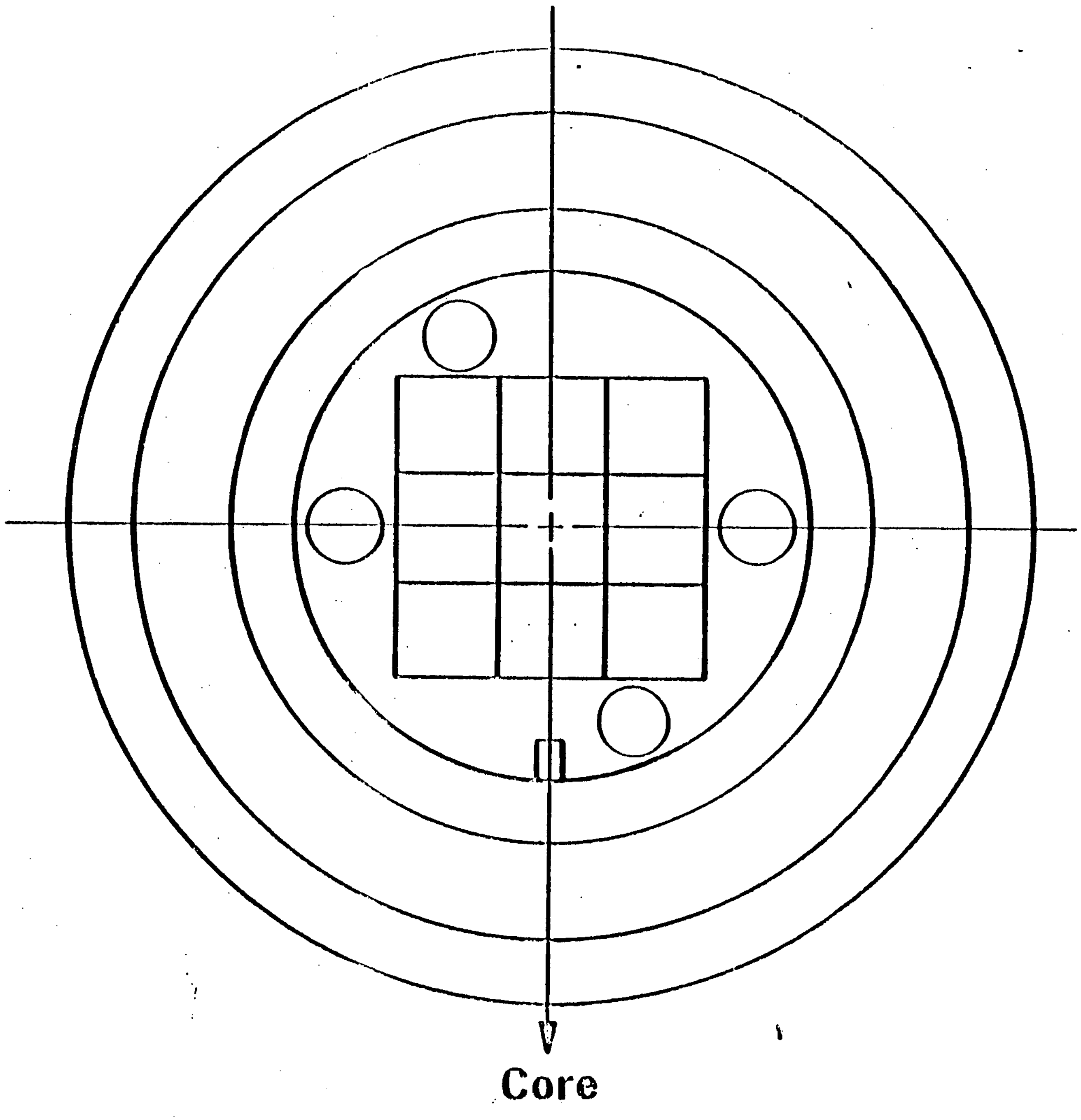
3. INTERNALS CONFIGURATION

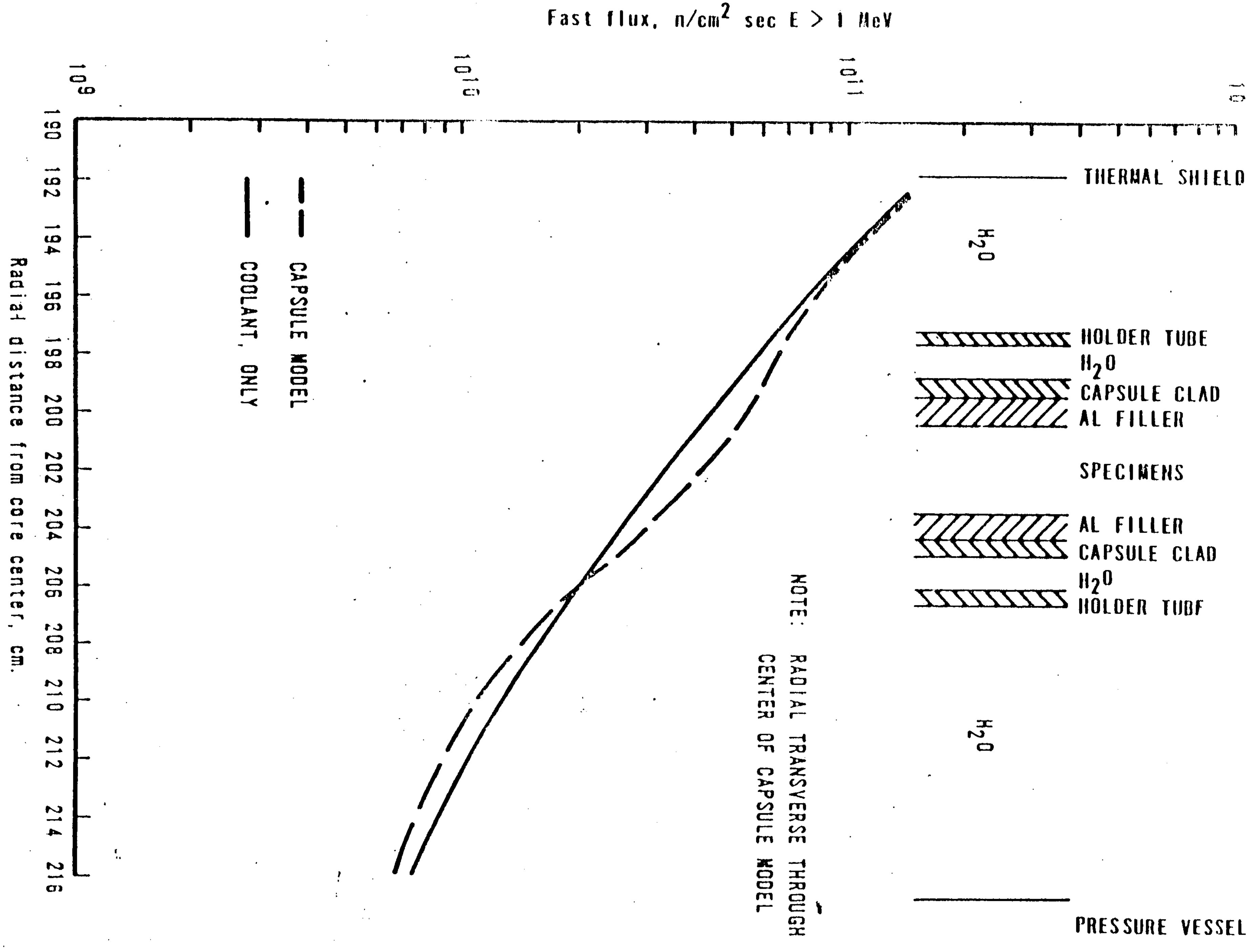
4. MATERIALS OF CONSTRUCTION

5. COOLANT DENSITY

6. POWER

7. POWER DISTRIBUTION


Core



Fast flux, n/cm² sec E > 1 MeV
10⁹
10¹⁰
10¹¹
10¹²
190
192
194
196
198
200
202
204
206
208
210
212
214
216
Radial distance from core center, cm.
CAPSULE MODEL
COOLANT, ONLY
THERMAL SHIELD
H₂O
HOLDER TUBE
H₂O
CAPSULE CLAD
AL FILLER
SPECIMENS
AL FILLER
CAPSULE CLAD
H₂O
HOLDER TUBE
H₂O
PRESSURE VESSEL
NOTE: RADIAL TRANSVERSE THROUGH CENTER OF CAPSULE MODEL

ESTIMATED UNCERTAINTY
IN FLUENCE VALUES

1. MATERIAL SPECIMENS IN CAPSULES, ± 14%

   SSC1 measurement comparison

   HEDL reevaluation of capsule data

2. VESSEL DURING CAPSULE EXPOSURE, ± 20%

   CAPSULE measurement

   CAPSULE location

   Radial extrapolation from capsule to vessel

   Azimuthal extrapolation to maximum location.

3. VESSEL AT EOL (MAX. LOCATION), ± 22%

   Vessel fluence from capsule measurement

   Time extrapolation to EOL

4. WELDS OUTSIDE BELTLINE REGIONS, ± 33%

   Maximum vessel location at EOL

   Spatial extrapolation in azimuthal and axial directions.

Oconee Plants Controlling Weld Metals

Sources of Irradiated Data

Plant	Weld I.D.	Res. Prog.		RVSP		Sim.W.W.	
		Cv	CF	Cv	CF	Cv	CF
OC-1	SA1229(c) SA1430(l)					X	X
OC-2	WF25(c)	X	X	X	X	X	X
	WF154(c)			X	X	X	X
OC-3	WF67	X	X				

Projected Oconee Plant Fluences

Plant	Weld I.D.	EOL Fluence I.S.	EOL Fluence T/4	T/4 Fluence* to 50 ft-lbs
OC-1	SA1229	9.4E18	5.2E18	8.9E18
	SA1585	1.2E19	6.8E18	4.9E19
	SA1493	9.0E18	5.0E18	1.5E19
	SA1430	1.1E19	6.1E18	9.7E18
OC-2	WF154	9.1E18	5.1E18	7.4E18
	WF25	1.2E19	6.7E18	1.1E19
OC-3	WF67	1.6E19	8.7E18	>2.0E19

* - Per BAW-1511P

Irradiation Data
Oconee Plant Weld Metals

		Fluence	USE ft-lbs	R.G. 1.99	
RVSP —	**WF25**	1.1E18	81/24	30%	(30%)
	WF154	9.0E18	88/39	44%	(42%)
		2.4E19	88/46	52%	(48%)
		7.2E18	72/20	28%	(41%)
		2.4E19	72/23	32%	(48%)
Res.Data —	**WF25**	3.9E18	78/10	13%	(37%)
		8.5E18	78/20	26%	(42%)
	WF67	3.6E18	74/7	9%	(32%)
		7.6E18	74/20	27%	(37%)
	SA1585	4.2E18	80/3	4%	(29%)
		9.5E18	80/17	21%	(35%)

Predicted Oconee Plant Δ Use

(Per BAW-1511P)

Plant	Weld I.D.	Fluence	Calendar Years
OC-1	SA1229(c)	> 5E18	> 16
	SA1585(c)	> 5E18	> 16
	SA1493(l)	> 5E18	> 16
	SA1430(l)	> 5E18	> 16
OC-2	WF154(c)	> 5E18	> 16
	WF25(c)	> 5E18	> 16
OC-3	WF67(c)	> 5E18	> 16

Surveillance Capsule Data

Observed Δ Use Vs Predicted Δ Use

Weld I.D.	Fluence	% Δ Use Drop Predicted*	% Δ Use Drop Observed
WF193	4.0E18	22	25
WF193	6.6E18	23	25
WF209-1A	1.0E18	13	21
WF209-1A	3.4E18	17	28
WF209-1B	8.2E17	12	12
WF209-1B	3.1E18	17	26
WF182	2.3E18	22	14

*Per BAW-1511P

RESOLUTION OF NRC CONCERNS

FOR OCII-A AND OCIII-B CAPSULES

PLANT SIMILARITY

Equivalent specimen damage per unit fluence.

Vessel fluence is accounted for in calculation.

CAPSULE ROTATION

No effect on specimen average fluence.

No effect on dosimetry average activity.

FLUENCE UNCERTAINTY

Estimated values are applicable.

USE LIMIT

Predictions in BAW 1511P of > 16 years (>13 EFPY) are still valid.

Based on current fluence analysis, USE is > 50 ft.-lbs. through 32 EFPY.

NRC PLANS WITH REGARD TO

FLUENCE UNCERTAINTY

What effect will quantification of fluence uncertainty have on vessel service life?

Will reported uncertainty values be treated individually or collectively?

CAVITY DOSIMETRY

Is there any NRC interest in the use of dosimeters in the cavity of B&W reactors?

SPECTRAL EFFECTS

Will DPA be used to evaluate vessel service life?