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FACIL:50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
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VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Responds to NRC 820316 ltr requesting addl info re pressurized thermal shock.

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TITLE: Thermal Shock to Reactor Vessel

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	NRR/DL/ORAB	11	1	0	NRR/DSI DIR		1	1
	NRR/DSI/RAB		1	1	NRR/DSI/RSB		1	1
	NRR/DST DIR		1	1	NRR/DST/GIB		1	1
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	RES VAGINS,M		1	1	RES/DET		1	1
	RES/DRA		1	1	RGN2		1	1
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FLORIDA POWER & LIGHT COMPANY

May 3, 1982
L-82-179

Office of Nuclear Reactor Regulation
Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Varga:

Re: Turkey Point Unit 4
Docket No. 50-251
Pressurized Thermal Shock

We have reviewed your letter dated March 16, 1982 requesting additional information regarding pressurized thermal shock. The following are our responses:

1. Provide the following information related to fluence determination:

(A) Plant specific information which would allow determination of the pressure vessel fluence. Such information should contain as built core and pressure vessel dimensions, regional material composition and neutron source for a two-dimensional (R-θ) and (R-Z) neutron transport solution, and

FPL Response

Refer to Attachment A for as built core and reactor pressure vessel dimensions. Refer to Attachment B for required material composition and neutron source.

(B) Plant specific values of the pressure vessel fluence and its estimated uncertainty.

FPL Response

Refer to FPL letter L-81-462 dated October 21, 1981 page one (1) of the attachment for specific values of fluence. Refer to WCAP-10019 page 41 for estimate of uncertainty.

2. Concerning Operator Action

(A) In your evaluation, the actions described do not provide the operator with clear direction for dealing with conflicting concerns that need to be evaluated when considering the operation of HPI and the charging flow as it relates to vessel integrity and

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maintaining core cooling. Provide an evaluation of the need and effectiveness of procedure modifications to clearly identify the concerns in the emergency operating procedures themselves. This should be done in contrast of depending upon upgrading operator training alone.

FPL Response

We have reevaluated our Turkey Point Unit Nos. 3 and 4 Emergency Operating Procedures using the criteria for procedure review presented in Section 2 of the NRC Staff audit report for the Carolina Power and Light Company's Robinson Unit No. 2 of April 15, 1982. Because the Turkey Point Emergency Operating Procedures are based on the Westinghouse Owners Group Procedure guidelines we conclude the procedural conclusions in this audit report would apply equally. Accordingly, we intend to revise our Emergency Operating Procedures to address the NRC Staff concerns regarding pressurized thermal shock by implementing a short term and long term program. The NRC concerns with the existing procedures will be resolved in our long term program by the resolution of TMI Action Item I.C. 1 with the Westinghouse Owner's Group both from a technical and a human factors standpoint. The short term program involves a modification and revision of our existing procedures to address the NRC Staff concerns that the existing Emergency Operating Procedures are weighted toward core cooling concerns and do not properly address vessel integrity. We specifically plan to reduce the 2000 psig safety injection termination criteria and to provide additional guidance to the operator for stabilizing temperatures in the reactor coolant system following a steamline break or feedwater line break (Loss of Secondary Coolant Accident).

(B) Provide a formal commitment to upgrade operator understanding of Pressurized Thermal Shock to the reactor pressure vessel.

FPL Response

Florida Power & Light Company commits to implementing an augmented training program focusing on the PTS issue with the emphasis on balancing the operator actions with regard to the safety systems to mitigate the consequences of transients and accidents. Presently classroom training concurrent with simulator training is being provided to licensed operators and shift technical advisors.

3. Concerning Input Data and Assumptions

3. Provide a description of the models or data used for:

(a) Heat sources (or sinks),
(b) Decay heat,
(c) ECC and feedwater temperatures (enthalpies) and flow rates,
(d) Primary and secondary relief capacities,
(e) Empirical correlation coefficients used for PTS evaluations,
(f) Operator Actions,
(g) Initial conditions.

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Pressurized Thermal Shock

FPL Response

Refer to WCAP 8945, May 1977, transmitted with our letter L-77-292 of September 14, 1977 and our letter L-82-26 of January 21, 1982 for Large LOCA and Large Steam Line Break assumption and conditions.

Refer to WCAP 10019 and our letter L-82-26 of January 21, 1982 for Small LOCA and Small Steam Line Break assumptions and conditions.

3.2 Provide a list of all transients or accidents by class (for example: excessive feedwater, operating transients which result from multiple failures including control system failures and/or operator error, steam line break and small break LOCA) which could lead to inside vessel fluid temperatures of 300° F or lower. Provide any Failure Modes and Effects Analyses (FMEAs) of control systems currently available or reference any such analyses already submitted. Estimate the frequency of occurrence of these events and provide the basis for the estimates. Discuss the assumptions made regarding reactor operator actions.

For a given initiating event, potential multiple and consequences failures need to be considered to identify those transients which could lead to a PTS problem.

FPL Response

This information will be provided directly to the NRC Staff by the Westinghouse Owners Group on May 20, 1982.

3.3 Identify all potential PTS events which have occurred at your facility. Include a designation of the operator actions and identify potential additional failures (including operator) which could have resulted in a more severe event.

FPL Response

We have reviewed Turkey Point Unit No. 4 records since initial operation using the following criteria for such PTS events:

1. Cooldown transient with reduced pressure and temperatures,

2. Duration of severe cooldown transient greater than 10 minutes,

3. Change of RCS temperature greater than 100°F, and

4. Conditions existed for repressurization following the cooldown transient.

Our review has revealed no PTS events that meet the above criteria.

Because the initiating events reviewed involved equipment failures and operator actions that did not lead to PTS events, it is our opinion that neither additional multiple failures nor personnel errors (of omission or commission) should be postulated.

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4. Concerning a Review of Operating History

Review your operating history at your plant and identify events which have resulted in exceeding the cooldown rate of 100°F/hr. as well as those events which could have exceeded the cooldown rate limit if not mitgated by plant controls or operator actions.

FPL Response

We have reviewed plant records and have found no events that exceed the cooldown rate of 100°F/hr.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems and Technology

REU/JEM/mbd

Attachment

cc: J.P. O'Reilly, Region II
Harold F. Reis, Esquire

Re: Turkey Point Unit 4
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ATTACHMENT A

We have reviewed our as built drawings and reference documents . We conclude that the dimension shown on Figure Nos. 1, 2, and 3 represent the Turkey Point Unit No. 4 Vessel.

Specifically, the eccentricity of the inner radius of the reactor pressure vessel shown on Figure 1, is no greater than 0.05 cm (0.011 in.). The surveillance capsule slots are less than 1° from the dimension shown.

(r,θ) REACTOR GEOMETRY

r_1 = 170.021 cm
r_2 = 175.182 cm
r_3 = 181.134 cm
r_4 = 187.959 cm
r_5 = 197.485 cm
r_6 = 217.488 cm
r_7 = 237.808 cm

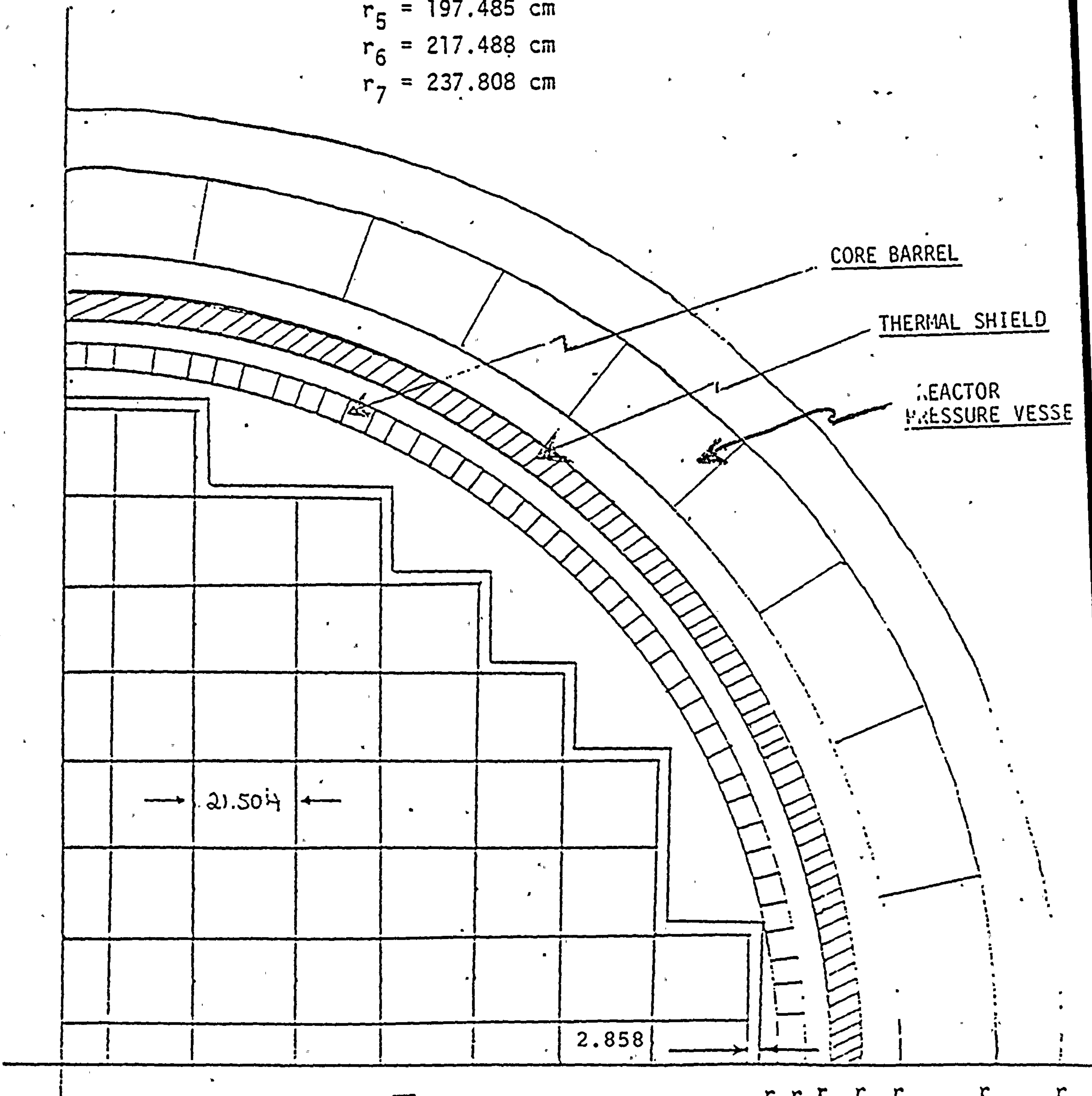

Figure 1

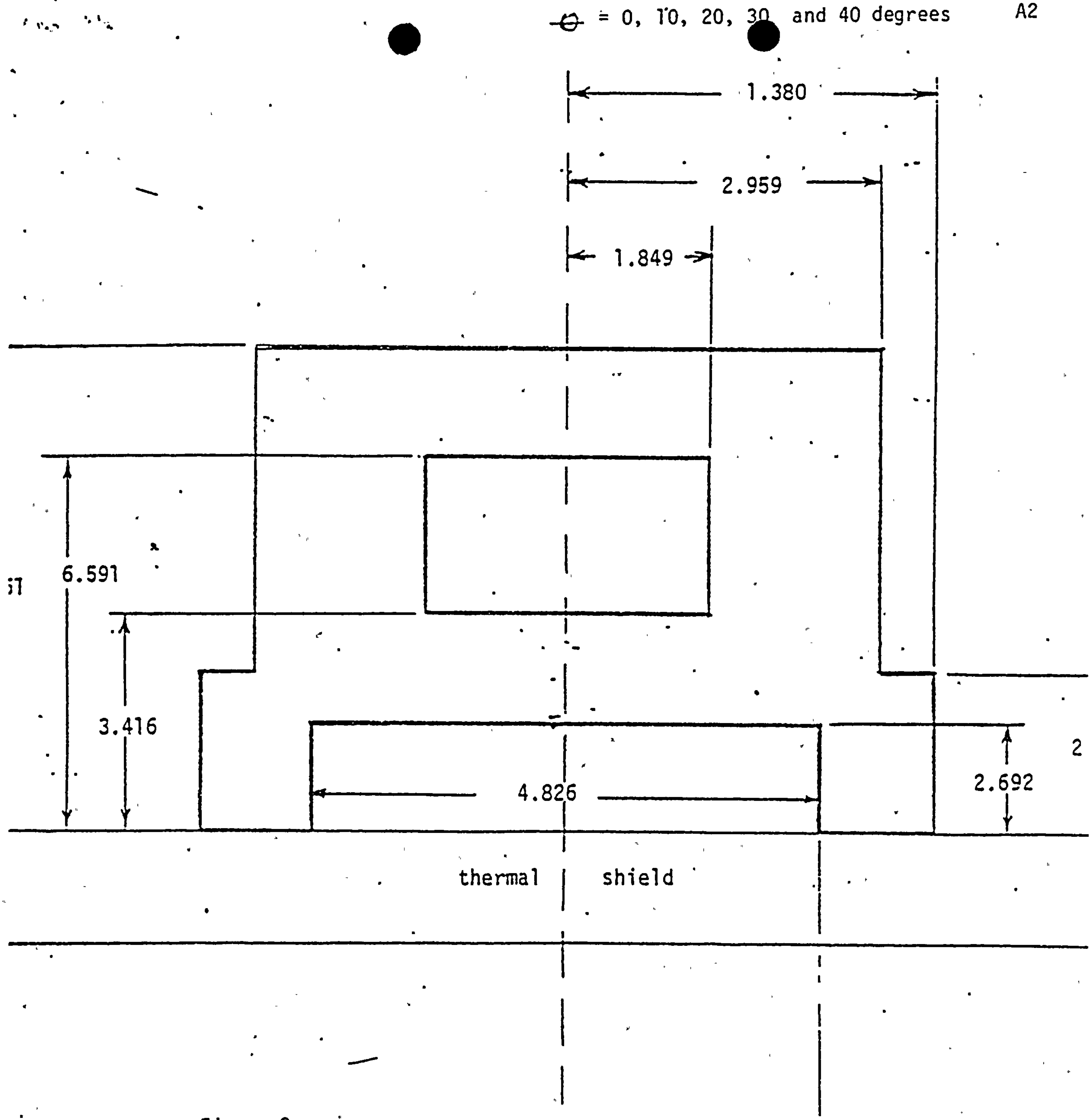

Figure 2

- Surveillance capsule dimensions (cm)

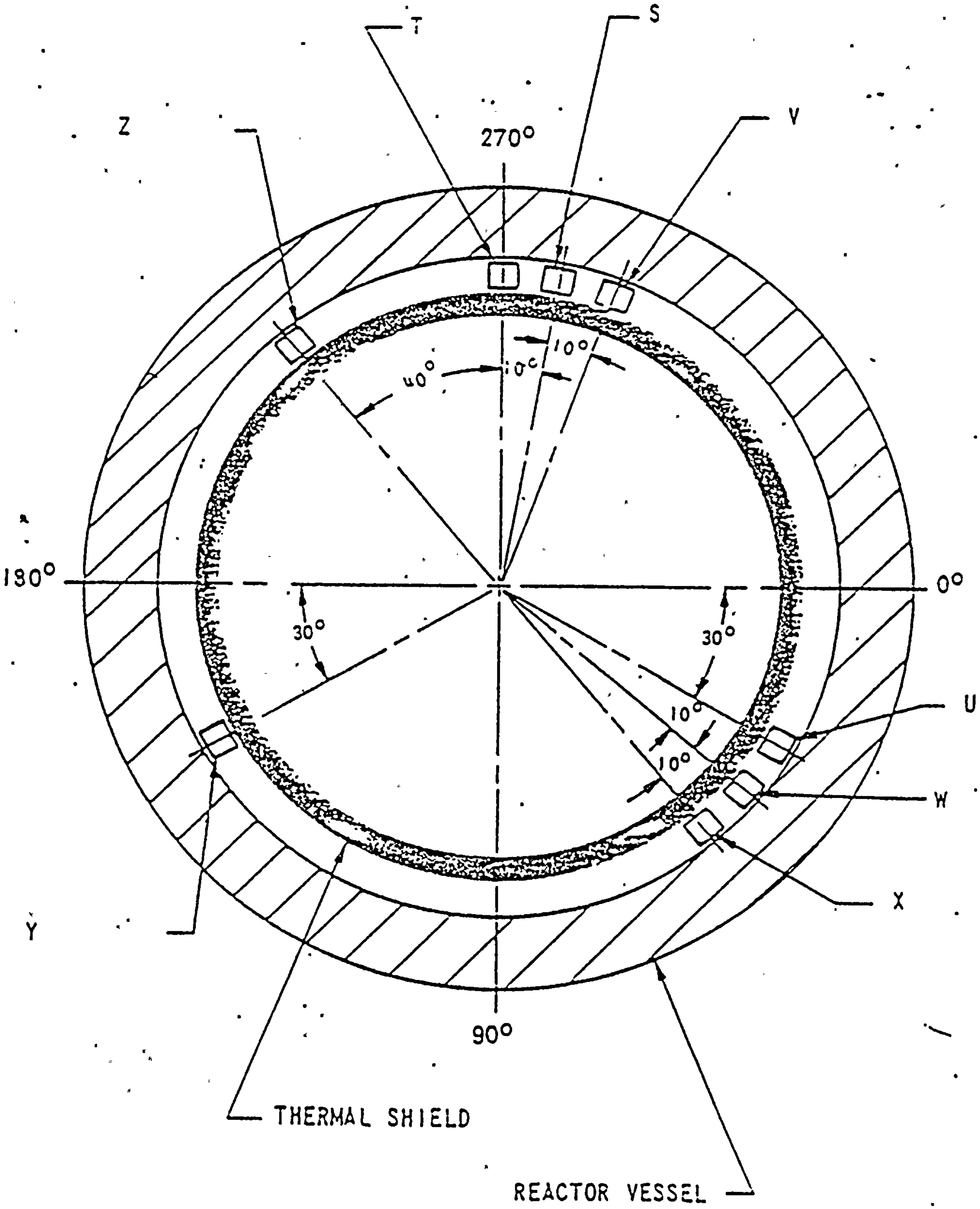

Figure 3
Arrangement of Surveillance Capsules

PLANT SPECIFIC FLUENCE DATA

Design basis reactor core power distributions which were used by Westinghouse in performing fast neutron fluence evaluations for the Turkey Point Unit 4 pressure vessel were supplied to the NRC in January 1982. That submittal consisted of the following items:

1. The material description of the reactor core presented in terms of volume fractions of solid material within a homogenized fuel zone.

2. Individual fuel assembly power density levels within one core octant. This data was presented relative to a core average of 1.0.

3. Spatial gradients of power density within each of the peripheral fuel assemblies. This data was presented relative to an assembly average of 1.0.

4. A time averaged relative axial distribution of power density applicable to long term fast neutron fluence projections.

The methodology used to define these long term design basis power distributions was outlined in WCAP 10019. Also described in WCAP 10019 was the methodology by which these power distributions were used in subsequent neutron transport calculations.

Following submittal of WCAP 10019 and the 150 day letters, the NRC requested additional plant specific information which would "allow determination of the current pressure vessel fluence". It is our understanding that this plant specific data is not intended to be used to perform fast neutron fluence projections for future operation. Therefore, the following information is provided for use by the NRC in their evaluation of the present condition of the pressure vessel.

The material composition submitted in the January letter was based on a fuel assembly design consisting of a 17 x 17 array of zirconium clad fuel rods. In actuality, the Turkey Point Unit 4 reactor employs a fuel design consisting of a 15 x 15 array of zirconium clad fuel rods. A comparison of the material volume fractions for a homogenized reactor core employing each of these fuel designs is given in Table 1. An examination of Table 1 shows that the compositions of the two fuel assemblies are quite similar and in our opinion the differences will have an insignificant impact on reactor vessel fluence calculations.

Plant specific peripheral assembly power distributions for cycles 1 through 7 are tabulated in Table 2. These data were extracted from the appropriate core design reports (WCAP's 7447, 8506, 8756, 8954, 9351, 9501, and 9826). Bias factors were applied to the design power distributions consistent with the methodology outlined in WCAP 10019. Also presented in Table 2 are the seven cycle time averaged power distributions for the peripheral assemblies. These average distributions were obtained by burnup weighting of the individual fuel cycle data sets. A comparison of the 7 cycle average data with the design

of Figure 1 shows that the plant specific power distribution will result in a somewhat lower fluence projection than that which would be calculated using the design basis distribution. It would appear that a reduction in pressure vessel fluence on the order of 20% might be realized. However, at this time neutron transport calculations using the plant specific power distributions have not been carried out. These computations must be completed before any reduction in the current pressure vessel fluence can be certified. It must also be re-emphasized that the plant specific data are applicable only for establishing the present condition of the pressure vessel. They should not be used to project forward in time.

An examination was also made of the variations in the power density gradients for the peripheral fuel assemblies at beginning of life and end of life for both 15 x 15 and 17 x 17 fuel rod arrays. The conclusion of this study was that these spatial gradients, relative to an assembly average power of 1.0, were quite similar in all cases examined. Therefore, the gradient information previously provided in the January submittal should also be used to generate plant specific fluence values for Turkey Point Unit 4. Likewise, the time averaged axial power distribution supplied in the January letter is suitable for the current analysis.

In the January 1982 letter, comparisons of calculated and measured fast neutron flux within reactor vessel surveillance capsules were also provided. Since that submittal, an error was identified in the Fe^{54} (n,p) Mn^{54} measurements. This error was caused by an improper calibration of the counting equipment used in the radiometric analysis. Updated comparisons are enclosed in this submittal in Table 3. The agreement between the updated measurements and the calculations is excellent.

FIGURE 1

COMPARISON OF TURKEY POINT UNIT 4 AND

DESIGN BASIS PERIPHERAL POWER DISTRIBUTIONS

Turkey Point/Design Basis

0.73 0.93	0.62 0.77				
	1.15 1.07	0.98 1.12	0.57 0.80		
			1.08 1.04	0.63 0.85	
				0.99 0.92	

Table 1

Material Composition of Reactor Core Region

Material	Volume Fraction	
	Design Basis	Turkey Point
Water	.58864	.5994
UO_2	.29967	.3062
Zirc - 4	.10035	.0842
Inconel - 718	.00281	
Stainless Steel	.00062	

Table 2

Turkey Point Unit 4 Peripheral Power

Assembly				Cycle				
No.	1	2	3	4	5	6	7	Avg.
2	.85	.79	.68	.53	.43	.81	.78	.73
3	.67	.65	.71	.65	.37	.59	.53	.62
4	1.08	1.18	1.17	1.22	1.17	1.11	1.21	1.15
5	.96	.95	1.11	.97	.68	.98	1.05	.98
6	.67	.66	.76	.43	.65	.44	.44	.57
7	.98	1.11	.97	1.18	1.17	1.08	1.12	1.08
8	.71	.67	.79	.82	.51	.47	.45	.83
9	.88	.78	.98	1.02	1.19	1.05	1.13	.99
Burnup	13900	8700	7400	9000	4250	13000	8850	

Note: The fuel assembly numbers refer to core positions designated in Figure 4 of the January 1982 submittal.

Table 3

Comparison of Measured and Calculated Fact Neutron Flux
for Surveillance Capsule S

Reaction and Axial Location	Radial Location (cm)	Adjusted Saturated Activity (dps/ms)	Fast Neutron Flux (n/cm^2 - sec) Capsule S	Calculated
$Fe^{54}(n.p)\ Mn^{54}$				
Top	191.46	5.76 (3)	1.32 (11)	
Middle	191.46	5.59 (3)	1.28 (11)	
Bottom	191.46	5.71 (3)	1.31 (11)	
Average			1.30 (11)	1.19 (11)
Top	192.46	6.21 (3)	1.43 (11)	
Middle	192.46	5.69 (3)	1.30 (11)	
Bottom	192.46	6.07 (3)	1.39 (11)	
Average			1.38 (11)	1.19 (11)

[illegible]