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CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Responds to request for addl info re 821208 proposed Tech Spec for plant heatup & cooldown curves based on results of capsule T.Info supplied by Westinghouse.Chart summary of neutron flux monitor encl.

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October 10, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Proposed Technical Specification
Heatup and Cooldown Curves
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

By letter dated December 8, 1982, we submitted a proposed Technical Specification for plant heatup and cooldown curves based on the results of Capsule T. The following information supplied by our contractor, Westinghouse, is provided in response to questions from members of the NRC Staff.

Question 1.

The use of an average generic power distribution instead of a plant specific distribution should be justified.

Response.

A key input parameter in the calculation of the integrated fast neutron exposure of the reactor vessel is the core power distribution. As stated in WCAP-10086, the neutron transport calculation for the R. E. Ginna reactor was based on a generic power distribution derived from statistical studies of long-term operation of Westinghouse 2-loop reactors.

It should be noted that the generic core power distribution is intended to serve a twofold purpose. First, a good relative neutron spectrum must be calculated within the surveillance capsule to enable an accurate evaluation of surveillance dosimetry. It must be emphasized that for this purpose only, the shape of the neutron energy spectrum is required; not the absolute magnitude. Second the results of

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the transport calculation must provide a vehicle for long-term (end-of-life) projection of vessel exposure. Here the absolute magnitude of the fast neutron flux at the pressure vessel is of paramount importance.

Clearly, to achieve the most direct comparison of analytical prediction with a measurement from an individual surveillance capsule the use of plant specific core power distributions may be more appropriate. However, since plant specific power distributions reflect only past operation or at best one cycle into the future, their use for long-term temporal projections cannot be justified; and the use of generic data which reflects long-term operation of similar cores may provide a more suitable approach.

The central question to be addressed in choosing a power distribution for the transport analysis is: Are the results of the discrete ordinates analysis expected to provide the best possible comparison with an individual measurement or are they expected to provide a reasonably accurate vehicle for predicting the condition of the pressure vessel for some future operating period?

In the Westinghouse analysis the emphasis has been placed on the latter goal of the transport calculation. The efficacy of this approach is demonstrated via the data comparison presented in answer to question 3.

Question 2.

The updating of the results of previous capsules R and V should be discussed and justified.

Response.

Dosimetry results were updated for capsules R and V for two basic reasons. First, in the capsule T analysis dosimetry results were gradient corrected in order to reference all measured data to the geometric center of the capsule. To provide a consistent data set for the R. E. Ginna reactor these gradient corrections were also applied to the prior measurements obtained from capsules R and V. Second, the neutron transport calculation associated with the capsule T analysis included improved modeling of the surveillance capsule and associated support structure. This improved transport model resulted in small changes in the neutron energy spectrum and, in turn, the spectrum averaged reaction cross-sections. Again, in order to provide a consistent data set, these improved reaction cross-sections were also applied

to the capsule R and V data. As a result of these updates, WCAP-10086 contains a set of fluence data that is referenced to a common location and is derived using the same nuclear parameters.

Question 3.

The benchmarking of discrete ordinates analysis procedures for these calculations should be established.

Response.

The most direct comparison of the results of neutron transport calculations with surveillance dosimetry measurements is achieved by examining the measured and calculated specific activities for each dosimeter reaction on an absolute basis. This type of comparison affords the opportunity to evaluate analytical predictions of both neutron flux magnitude and to some extent neutron energy spectra at the surveillance capsule locations.

Measured specific activities of fast neutron dosimeters removed from 15 2-loop plant surveillance capsules are summarized in Table 1. This data base spans six reactors with as many as three capsules removed from a single reactor. The irradiation times associated with the data base range from a low of 1.25 to a high of 6.81 effective full power years (EFPY). Ten of the capsules were extracted from the 13° position, four from the 23° position, and a single capsule from the 33° position.

To facilitate experimental/analytical comparisons, all of the measurements have been normalized to a reactor power of 1961 Mwt. In addition data from the 23° and 33° capsule positions have been normalized to the 13° location using analytically developed gradient information. The U-238 (n,f) Cs-137 measurements listed in Table 1 have been adjusted to account for U-235 impurity in the dosimeter and for Pu-239 buildup over the course of the irradiation. However, no adjustment has been made to account for photofission. Preliminary calculations indicate that for a 2-loop plant, surveillance capsules photofission effects are on the order of 1 - 2 percent.

The following observations can be extracted from the data listed in Table 1:

1) For all five fast neutron reactions the data set is remarkably consistent. The bounds on the measured specific activities range from a low of ± 9 percent for the Fe-54 (n,p) Mn-54 reaction to a high of ± 22 percent for the U-238 (n,f) Cs-137 reaction.

2) There appears to be no discernible bias in the experimental data either from reactor to reactor or from year to year for a given reactor. In fact, the variation in the data sets could be attributed to experimental uncertainty.

3) The discrete ordinates calculations using the generic power distribution, generic reactor dimensions, and P_1 scattering approximation are in excellent agreement with the measured data. With the exception of the Cu-63 reaction, calculations agree with the average measured reaction rates within ±10 percent. The Cu-63 calculation lies within 20 percent of the measured reaction rate and much of this discrepancy can be attributed to the dosimetry cross-section rather than the calculated neutron flux level.

From these observations, it can be concluded that the use of the methodology described in WCAP-10086 is justifiable based on the consistency of the data set and on the good agreement between analysis and measurement. For reactors using a predominantly out-in fuel loading pattern the generic analysis is shown to be reasonably accurate for prediction of individual capsule fluence levels as well as for prediction of long term trends. If low leakage loading patterns were to be employed, analytical projections of neutron fluence would be conservative by varying degrees depending on the specific fuel management scheme employed.

The fact that the C/E ratios are similar for most reactions listed in Table 1 implies that the discrete ordinates calculations are doing an adequate job not only at predicting neutron flux magnitude but also fast neutron energy spectra within the surveillance capsules.

Question 4.

An analysis for the error and uncertainty bounds should be provided.

Response.

The data and calculations presented in answer to question 3 indicate that analytical predictions at the surveillance capsule locations agree with average measurements to within ±20 percent; actually ±7 percent if the Cu-63 reaction is deleted. Further, the measured data is bounded by a range of from ±9 percent for the Fe-54 reaction to ±27 percent for the

U-238 reaction. Note that these are bounds not 1σ levels. Thus, the agreement between calculation and measurement falls well within the bounds of the measured data. Based on this comparison an uncertainty of ±10 percent 1σ is assigned to the calculation at the surveillance capsule location. To account for uncertainties in extrapolating neutron flux levels from the surveillance location to the pressure vessel, an additional uncertainty of ±15 percent 1σ is added. Thus, calculated flux and fluence levels within the pressure vessel are anticipated to have an associated uncertainty of ±25 percent at the 1σ level.

It should be recognized that implementation of low leakage core loading patterns would tend to make calculations conservative relative to actual flux levels.

Question 5.

The use of the P_1 approximation should be justified.

Response.

The benchmarking comparisons given in answer to question 3 reflect the use of P_1 scattering cross-sections in the overall methodology. Recent calculations using a P_3 expansion with all other input parameters held constant result in analytical predictions that exceed the P_1 results by 13 percent at the surveillance location. Thus, the P_3 are conservative by 7 - 22 percent and in the case of Fe-54 and Ni-58 represent upper bounds relative to the measurements.

Very truly yours,

John E. Maier

John E. Maier

Summary of Neutron Flux Monitor Saturated Activities at the Center of 2-Loop Plant Surveillance Capsules - 1961 Mwt - Normalized to the 13° Location

Plant/ Capsule	Capsule Location	Operating Time (EFPY)	Monitor Activity (dps/GM) $Fe^{54}(n,p)Mn^{54}$	$Ni^{58}(n,p)Co^{58}$	$Cu^{63}(N,\ )Co^{60}$	$Np^{237}(n,f)Cs^{137}$	$U_{238}(n,f)Cs^{137}$
A-1	13°	1.49	6.25×10^6	8.35×10^7	--	8.40×10^7	7.19×10^6
A-2	33°	3.60	6.37×10^6	8.65×10^7	5.50×10^5	9.29×10^7	8.60×10^6
A-3	13°	5.08	7.05×10^6	1.06×10^8	5.98×10^5	8.28×10^7	9.24×10^6
B-1	13°	1.52	6.85×10^6	7.96×10^7	5.36×10^5	8.50×10^7	1.05×10^7
B-2	23°	3.45	6.48×10^6	9.63×10^7	5.33×10^5	8.62×10^7	8.78×10^6
B-3	13°	5.10	6.34×10^6	1.02×10^8	5.68×10^5	8.77×10^7	9.82×10^6
C-1	13°	1.64	6.89×10^6	1.12×10^8	5.21×10^5	--	6.88×10^6
C-2	13°	2.55	6.33×10^6	8.08×10^7	5.64×10^5	9.33×10^7	8.16×10^6
C-3	23°	6.81	6.03×10^6	8.33×10^7	6.27×10^5	9.96×10^7	9.21×10^6
D-1	13°	1.25	6.33×10^6	8.85×10^7	5.50×10^5	9.32×10^7	9.33×10^6
D-2	13°	4.50	--	9.98×10^7	5.68×10^5	--	8.37×10^6
E-1	13°	1.34	6.01×10^6	8.69×10^7	5.43×10^5	7.26×10^7	8.24×10^6
E-2	23°	4.59	6.54×10^6	1.10×10^8	6.19×10^5	9.25×10^7	9.24×10^6
F-1	13°	1.28	6.70×10^6	1.03×10^8	5.86×10^5	9.37×10^7	9.59×10^6
F-2	23°	3.99	6.48×10^6	1.01×10^8	5.60×10^5	9.96×10^7	9.18×10^6
Measured	Max/Min		1.17	1.41	1.20	1.37	1.53
Measured	Average		6.48×10^6	9.46×10^7	5.66×10^5	8.95×10^7	8.82×10^6
	Calculated		6.71×10^6	9.93×10^7	4.58×10^5	8.32×10^7	9.16×10^6
	Calculated/Measured		1.04	1.05	0.81	0.93	1.04