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AUTH.NAME AUTHOR AFFILIATION
TUCKER,H.B. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director
STOLZ,J.F. Operating Reactors Branch 4

SUBJECT: Forwards summary of 830512 meeting w/util & B&W in response to 830327 request for addl info reactor vessel surveillance capsule dosimetry results contained in BAW-1697 & 1699.

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DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 27, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

By letter dated March 27, 1983, the NRC Staff requested additional information relative to the Oconee reactor vessel surveillance capsule dosimetry results provided in BAW-1697 and BAW-1699. On May 12, 1983, Duke and B&W made a presentation to the Staff in response to this request.

Attached please find a summary of this meeting which constitutes formal response to the NRC request for information.

Very truly yours,

Hal B. Tucker

RLG/php
Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

Mr. J. C. Bryant
NRC Resident Inspector
Oconee Nuclear Station

Mr. John F. Suermann
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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SUMMARY OF NRC/DUKE POWER/B&W MEETING

May 12, 1983

The following is a discussion of the NRC concerns expressed in the March 22, 1983 letter to Duke Power Company relative to analysis of the OCII-A and OCIII-B capsules. This discussion follows the line of reasoning that was presented, and apparently accepted, at the May 12 meeting in Bethesda. However, we wish to restate our concern with the apparent acceptance and reliance on the fluence uncertainty values presented herein. Not only are the data considerably less than rigorous, but insufficient effort has been made to even define all the pertinent pheonmena. For example, no distinction has been made between reactors with and without capsule dosimetry; nor has the potential benefit of redundant flux wires been considered. In our opinion, if fluence uncertainties are going to significantly affect vessel life, the present set of values should be reevaluated.

Plant Similarity

Operation of the Integrated Reactor Vessel Surveillance Program in B&W 177 FA reactors was approved by the NRC (V. Stello Jr., to W. O. Parker - date 7/14/77). Material specimens from guest reactors are irradiated in host reactors to obtain material property data, and vessel fluence calculations for guest reactors are normalized to capsule fluence calculations in host reactors. The basis for this procedure is the similarity of configuration and operating conditions between all B&W 177 FA reactors. For specimen irradiation data to be valid for these reactors, material property change per unit fluence must be the same. Property change per unit fluence is dependent only on the relative energy spectrum of the flux, which in turn, is dependent on core shape, core composition and fission spectrum in peripheral fuel assemblies, configuration of internals components, and inlet coolant temperature. These quantities are essentially identical for the 177 FA reactor design. Thus, material specimen data obtained in host reactors are applicable to guest reactors. Vessel fluence, in addition to being dependent on the same phenomena as the material specimens, is also dependent on the magnitude of the fast flux. Since plant specific models are used for all vessel fluence calculations, flux magnitude is accounted for in the power distribution input. These power distributions are based on PDQ criticality calculations and are subsequently verified by comparison to in-core measurements. Vessel fluence calculations for guest reactors are normalized to capsule calculations for host reactors. Because of design similarity, the same calculational model, except for power distribution input and inclusion of the capsule, is used for both calculations. Thus, only errors associated with modelling the power distribution could differ between the two calculations, and that is unlikely because the same procedure is used for both power distributions. Therefore, the capsule normalization factor, which is used to zero out errors associated with analytical modelling, is applicable to both the host reactor and the guest reactor calculations. We recognize the possibility that the relatively greater number of fissions in 239_{Pu} in low leakage core designs could affect intra-reactor comparisons. Although not considered to be of immediate concern (B&W reactors have only recently started using the low leakage fuel cycle), the 239_{Pu} spectral affect is scheduled for investigation.

Capsule Rotation

Due to problems with the mechanical integrity of surveillance capsule holder tubes, the holder tubes were redesigned prior to installation in the host reactors. To fit into the new holder tubes, new end caps were welded to each capsule. Because external markings on the capsules were obscure, in some cases the new end caps were not aligned so that the capsule keyway faced the reactor core. Thus, irradiation subsequent to cycle 1 exposure in the original holder tube design may occur with capsules in a different rotational orientation. This situation does not affect the fluence analysis presented for the OCII-A and OCII-B capsules. Fluence data used to correlate material property changes is unaffected because the fluence value at the capsule center is used to represent all specimens. Fluence at the capsule center is the same regardless of rotational position. Data scatter in the material property change correlations is sufficiently great that the use of the fluence gradient between Charpy specimens is not warranted. Use of dosimetry data is also unaffected because for each dosimeter reaction an average measured activity from the four dosimeter strings is compared to a calculated value at the capsule center to obtain the normalization factor. The four dosimeter strings, each containing 6 flux wires, are located symmetrically around the capsule center. Regardless of rotational orientation, this symmetry exists. The linear averaging of activities was shown to be valid by comparison to within 1 or 2% averages based on calculated flux gradients for several rotational positions.

Fluence Uncertainty

To date, only nominal values of fluence have been included in surveillance capsule reports because uncertainty factors were neither required nor used by the NRC. For similar reasons, little effort has been expended in the past in quantifying the overall uncertainty of in-reactor fluence calculations. However, a set of values, which differentiate between the various uses of fluence data, has evolved over several years of experience within the capsule analysis program. These data are listed in the March 22, 1983 NRC letter and are considered applicable to reported fluences. Please note that the fourth category of 33% pertains to welds outside the beltline. The uncertainty ranges are based on comparisons to integral experiments (PCA, HEDL reevaluation, in-reactor capsule dosimetry), uncertainty of fundamental input data (cross sections, fission yield, isotope half life), uncertainties in reactor conditions and configurations (dimensional tolerance, material composition, coolant temperature), and a considerable amount of engineering judgement. To obtain overall uncertainties, values associated with each component of a calculation were combined using the square root of the sum of the squares technique which implies independent behavior of each variable. Although quantified uncertainties were used when available, it was necessary to estimate many of the effects.

Estimate to Reach 50 Ft. lbs.

The effect of irradiation on the drop in the upper shelf energy of the weld metal in the Oconee reactor pressure vessels has been addressed as a part of Phase I of the B&W Owners Group Program for Evaluation of Reactor Vessel Material Properties as described in BAW-1474, December, 1977. As a result of this program, an evaluation was made of the behavior of upper shelf energy as influenced by irradiation and chemical composition of submerged arc weld metals.

From this study, a correlation was developed as reported in BAW-1511P, October, 1980. This correlation was used to determine that the limiting belt-line weld metals would not reach the 50 ft-lb level until at least 16 calendar years of operation.

Since these original estimates were made, the fluence on the reactor vessel has been reduced by changing the type of fuel cycle. This change has reduced the end of life fluence on the vessel by nearly a factor or two. This would indicate that the current estimate for reaching the 50 ft-lb level will be in excess of 15 EFPYs.

The prediction of materials behavior as it is influenced by irradiation is still a developing science and, therefore, requires that any correlation be updated or improved as more data or information is obtained. The currently used B&W correlation for predicting the drop of upper shelf energy is in the process of being updated as a result of the new data developed since the original correlation was produced. Once this updated correlation is available, new projections will be made as to the most probable time that each unit will be expected to reach the 50 ft-lb level.