SAFETY EVALUATION BY THE ENGINEERING BRANCH OF OPERATIONAL TECHNOLOGY

SUPPORTING AMENDMENT NO. TO LICENSE NOS. DPR-38, 47 and 55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION - UNITS 1, 2 AND 3

DOCKET NOS. 50-269, -270, AND -287

Introduction

The original Oconee design included three reactor vessel surveillance specimen holder tubes (SSHTs) located near the reactor inside vessel wall. Each of these SSHTs housed two capsules containing reactor vessel surveillance specimens. When failures of the SSHTs occurred at other Babcock & Wilcox (B&W) designed plants, the licensee shut down the three Oconee facilities in succession, starting in March, 1976 to inspect the SSHTs. The inspection revealed that all of the SSHTs had suffered some damage. To prevent further damage all surveillance capsules and all parts of the SSHTs that had failed or were deemed likely to fail during the remainder of that operating cycle were removed from the vessels.

Proposed Program

Since the discovery of the damage to the SSHTs, B&W has undertaken the design, manufacture and testing of an improved SSHT. SSHTs of this improved design are presently installed in Davis-Besse 1, Crystal River-3 and Three Mile Island-2. All three of these plants have reactors supplied by B&W and all are in the process of beginning initial operation at the present time or within the next few months. In addition, all of these reactors have the same basic B&W 177 fuel assembly design as Oconee 1, 2 and 3. The acceptability of the redesigned SSHTs has been demonstrated by a test program reviewed and approved by the staff and conducted in conjunction with the Hot Functional Test performed at Davis-Besse 1.

Installation of the redesigned SSHTs in the Davis Besse 1, Crystal River-3 and TMI-2 reactor vessels did not present any unusual difficulties because it was performed prior to neutron activation of the reactor internals. Studies of methods to install the redesigned SSHTs in the irradiated B&W reactors indicate that substantial difficulties will be experienced--primarily because precision machining, alignment and inspection must be performed remotely and under water. Although such problems do not in themselves justify relief from a requirement to re-install the SSHTs in Oconee 1, 2 and 3, they would cause significant radiation to personnel. Based on their experience in removing the SSHTs at Three Mile Island-1 and Rancho Seco-1, B&W estimated that installing SSHTs in irradiated reactors would result in personnel exposure of about 100 man-rem per reactor. In the interest of maintaining the radiation exposure of plant personnel as low as reasonably achievable, the licensee, in cooperation with B&W and the owners of other B&W 177 fuel assembly plants, has proposed an alternative program that does not require reinstalling the SSHTs in Oconee 1, 2 and 3 and the other irradiated B&W plants.

This program is very complex, as it includes provisions to provide additional information, if required under Appendix G 10 CFR 50 Paragraph V.C., in addition to the normal requirements of Appendix H.

The proposed plan involves integrating the interrupted surveillance programs into the programs for new plants in a manner generally similar to that covered in Appendix H, 10 CFR 50, paragraph II.C.4, except that the plants are at different sites. There are three distinct features of these proposed programs:

1. A host-reactor feature, in which the original surveillance materials from one or more reactors that have been in service will now be irradiated in a new host reactor, that can be fitted with the newly-designed capsule holders on the thermal shield in less time and without radiation exposure of the workmen, and

2. An augmented surveillance feature in which there will be more weld metal specimens and some larger fracture mechanics (compact tension or CT) specimens placed in the capsules, and

3. A data-sharing feature in which all available irradiation data for all of the beltline welds of a given reactor will be considered in predicting its adjusted reference temperature and in making any fracture analyses for that reactor. Typically, several of the welds in any one vessel were made with the same weld wire and flux as those used on some other reactors. The data sharing feature is required because the welds in these reactors have high radiation sensitivity due to high copper content, large and random variation of copper from point to point in the weld, and low initial upper shelf energy.

The specific program proposed for Oconee 1, 2 and 3 involves installing the original Oconee surveillance capsules in extra locations provided in the Crystal River 3 vessel. This plan will accomplish the original purpose of obtaining information on the effect of radiation on material that is representative of (although not identical to) the material in the Oconee reactor vessels on a schedule that provides an appropriate lead time over the vessel irradiation rate. The overall integrated program also will provide information from surveillance programs in Crystal River 3, Three Mile Island 2, and Davis Besse 1 on material considered to be essentially identical to the actual welds in the Oconee vessels. It is also important to note that still more information relevant to the Oconee vessel materials will be obtained from the NRC funded HSST irradiation programs underway. Details are provided below.

Oconee 1

There are two weld materials of primary interest for the Oconee 1 vessel, Procedure Qualification (P.Q.) numbers* SA-1229 and SA-1585. These are

*Weld materaials are specifically identified by the ASME Code by the procedure Qualification Test number. A procedure qualification test is required on each combination of heat of weld wire and batch of flux.

used in the I.D. portion of the upper circumferential weld, and in the center circumferential weld respectively. The end of life (EOL) fluence for both of these welds is estimated to be 1.2×10^{19} nvt, and both have compositions that are expected to make them relatively sensitive to radiation damage. Weld P. Q. No. WF-25, used for the O.D. portion of the upper circumferential weld is radiation sensitive, but fluence at the weld is reduced by attenuation through the wall. Weld P.Q. Nos. SA-1073, SA-1493 and SA-1430, used for the longitudinal welds, have lower copper, making them less sensitive to radiation. Further, the EOL fluence at the azimuthal locations of these longitudinal welds is lower (0.7 to 1.0×10^{19}) so they will not become limiting during the service life. The lower circumferential weld will not be limiting, because the fluence at this location is estimated to be at least an order of magnitude lower than that of the other circumferential welds.

The original surveillance material, WF-112, was made using the same heat of filler wire but a different batch of flux as WF-154, one of the controlling welds in Oconee 2. Metallurgical considerations suggest that the radiation behavior is affected more by the wire than the flux, thus WF-112 is expected to respond to radiation much like WF-154. In addition, WF-112 data will be pertinent to Arkansas Nuclear One, Unit 1. In fact, the results of capsule E have already been reported. Also, the data will be a useful part of the data base for B&W vessels.

Table 1 shows where samples of the pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available under the present plan. The irradiation schedule and withdrawal dates shown will be modified to optimize the information obtained as indicated to be appropriate as initial test results are obtained and evaluated.

Oconee 2

There are only two weld materials of interest for the Oconee 2 vessel, P.Q. numbers WF-25 and WF-154. They are used in the center and upper circumferential welds. Fluence is expected to be the same at the two welds, but the higher copper content of WF-25 means that it is predicted to be more radiation sensitive. Fluence at the lower circumferential weld, WF 112, is too low for it to be of concern. There are no longitudinal welds in this vessel beltline.

The original surveillance material, WF-209-1, while not identical to any of the beltline welds in B&W reactors, is predicted to be radiation sensitive, based on its copper and phosphorus contents. Data from WF 209-1 will be a useful addition to the data base for these reactors.

Table 2 shows where samples of the pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available under the present plan.

Oconee 3

There are three weld materials of primary interest for the Oconee 3 vessel, Procedure Qualification (P.Q.) numbers* WF-67 and WF-70 in the center circumferential weld, and WF 200 in the upper circumferential weld. The end of life (EOL) fluence for both of these welds is estimated to be 1.2×10^{19} nvt, and the three weld materials have compositions that are expected to make them relatively sensitive to radiation damage.

Another shell weld, the lower circumferential, is made of a material that is expected to be radiation sensitive (P.Q. No. WF-169), but the EOL fluence at this location is estimated to be at least an order of magnitude lower than that of the other circumferential welds, so it will never be limiting.

The original surveillance material, WF 209-1, is the same as that used in Oconee 2. The weld wire used for WF 209-1 was also used for WF-70, one of the controlling welds in Oconee 3.

Table 3 shows where samples of the pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available under the present plan.

In addition to this integrated program, "research" capsules containing tensile, Cv, and several sizes of CT specimens of B&W archive material will be included in the overall B&W power reactor surveillance program. These are shown as "capsule R-1 and R-2" in Tables 1, 2 and 3. For example, samples of the weld most likely to be limiting in Oconee 1, SA-1585, will be irradiated in Crystal River 3, and samples of a weld made of the same heat of weld wire as WF-154 will be irradiated in the Davis Besse program. Details of withdrawal schedules will be reviewed again later, and will depend on test results from the other programs.

Research programs being funded by the NRC will also provide continual information on the effect of radiation on these specific weld materials and on several additional B&W weld materials expected to respond to radiation in a similar manner. These programs, HSST-2 and HSST-3, consist of many tensile, Cv, and CT specimens irradiated in a test reactor. Although information on shift in RT_{NDT} will be obtained, the main emphasis of the HSST programs is to develop methods that can be used to better evaluate low shelf toughness using the rather small specimens used in the power reactor programs.

Staff Evaluation

The staff has evaluated the effectiveness of this overall program plan, and has concluded that the information to be developed that is directly and indirectly relevant to the Oconee reactor vessels will be sufficient to provide assurance of safety margins against vessel failure that comply with Appendix G, 10 CFR 50. Further, it is the staff's opinion that even without additional irradiation surveillance programs in the Oconee vessels,

TABLE 1

RADIATION DATA FOR

OCONEE - 1 REACTOR

Weld	Capsule	Reactor	Removal Date	Specimen Types
WF-25 (Upper Circumferential, 39%, on OD)	TMI-1E	TMI-1	1976	Cv, Tensiles
	TMI-1A	TMI-2	1982	Cv, Tensiles
	TMI-1C	TMI-2	1990	Cv, Tensiles
	R-1	TMI-2	1982	Cv, CT
	R-2	TMI-2	1989	Cv, CT
	HSST-2	Test Reactor	1977-78	Cv, CT to
	HSST-3	Test Reactor	1978	4.0T
	NRL	Test Reactor	1977-78	Cv,CT
	NRL	Test Reactor	1977-78	Cv,CT
SA-1229 (Upper Circumferential, 61% on ID)	none			
SA-1585 (Center Circumferential)	R-1	CR-3	1982	Cv, CT
	R-2	CR-3	1989	Cv CT
	HSST-3	Test Reactor	1978	Cv, CT to 4.0T
SA-1073 (Longitudinal, in upper shell course)	none			
SA-1493 (Longitudinal, in middle shell course)	none			
SA-1430 (Longitudinal, in lower shell course)	none			
SA-1135 (Top Circumferential Weld) WF-9 (Lower Circumferential Weld)				

TABLE 2

RADIATION DATA FOR

OCONEE -2 REACTOR

Weld	Capsule	Reactor	Removal Date	Specimen Types
WF-25 (Center Circumferential Weld)	TMI-1E	TMI-1	1976	Cv, tensile
	TMI-1A	TMI-2	1982	Cv, tensiles
	TMI-1C	TMI-2	1990	Cv, tensile
	R-1	TMI-2	1982	Cv, CT
	R-2	TMI-2	1989	Cv, CT
	HSST-2	Test Reactor	1977-78	Cv, CT to 4.0T
	HSST-3	Test Reactor	1978	
	NRL	Test Reactor	1977-78	Cv, CT
	NRL	Test Reactor	1977-78	Cv, CT
WF-154 (Upper Circumferential Weld)	none, but has same weld wire as WF-112.			
WF-112 (Lower Circumferential Weld)	OCI-E	Oconee 1	1977	Cv, tensile
	OCI-A	Crystal R.3	1985	Cv, tensile
	OCI-C	Crystal R.3	1989	Cv, tensile
	R2	Davis-Besse	1989	Cv, CT

TABLE 3

RADIATION DATA FOR

OCONEE - 3 REACTOR

Weld	Capsule	Reactor	Removal Date	Specimen Types
WF-67 (Center Circumferential Weld) 75%, I.D.	R-1 R-2 R-1 R-2 NRL	Davis Besse -1 Davis Besse-1 Crystal River 3 Crystal River 3 Test Reactor	1981 1989 1982 1989 1977-78	Cv,CT Cv,CT Cv,CT Cv,CT Cv, CT
WF-70 (Center Circumferential Weld) 25%, O.D.	R-1 R-2 R-1 R-2 HSST-3	Davis-Besse 1 Davis-Besse 1 Crystal River 3 Crystal River 3 Test Reactor	1981 1989 1982 1989 1978	Cv,CT Cv,CT Cv,CT Cv,CT Cv,CT to 1.6T
WF-200 (Upper Circumferential Weld)	none			
WF-169 (Lower Circumferential Weld)	none			

the proposed program will provide more useful information than would have been obtained from the original surveillance program.

Until data become available from the surveillance program, a conservative prediction of radiation damage can be made by using R.G. 1.99* for at least the next 5 years of operation. This Regulatory Guide is based on the staff's analysis of all data available at the time the Guide was written. New data, in particular the results of the augmented integrated surveillance program described above, will be used to update the Guide periodically. Predictions of the adjustment of reference temperature and the drop in upper shelf energy are given graphically as functions of copper and phosphorus content and of fluence. In addition, there is an "Upper Limit" line on each graph, which is to be used when information about the copper and phosphorus contents is inadequate. Because the chemical analyses of the B&W welds have shown considerable variation, the staff intends to use the Upper Limit lines as the basis for any prediction required at this time.

The staff also has considered the uncertainties involved in applying radiation effects information obtained in other reactors to the Oconee vessels. The major uncertainties involved are:

1. Accuracy of neutron fluence calculations
2. Magnitude and effect of variation in neutron spectra between reactors
3. Magnitude and effect of variations in irradiation temperature between reactors.
4. Magnitude and effect of variations in rate of irradiation on material properties.

The effects of these variables have been studied for at least 20 years. Although some uncertainties still remain, the effects are fairly well established and understood as discussed below.

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*Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", April 1977.

1. Neutron flux calculations for the reactor vessel wall and irradiation capsule locations have been developed over many years. The dosimetry used in irradiation capsules has furnished information that was used to check out and refine the calculational methods. It is generally believed that the fast neutron flux and fluence in these locations can be calculated to an accuracy of $\pm$ 20%, particularly if some dosimetry checks are available. Dosimeters from the original Oconee surveillance program were removed and tested, so the fluence calculations for the vessel can be verified.

   In this connection, it should be emphasized that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence, so uncertainties of 20 to 50% in fluence are not highly significant.

   The staff has also considered the fact that the design of the Oconee vessels, internals, and cores is almost identical to that of the other reactors that will be used to obtain radiation effects information.

   These considerations are the basis for the staff's conclusion that uncertainties in the calculation of neutron fluence will be small, and the effect of such uncertainties on the assessment of the radiation effects on the vessel material will also be small.

2. Although differences in neutron energy spectra can cause uncertainties in the effects of radiation on material when this is evaluated without considering spectrum effects, only very large differences in spectra are significant. The variations from one B&W reactor to another are claimed to be relatively minor, because they have similar geometry.

   The staff considered the possible differences in neutron spectra that could occur between the B&W power reactors involved in the integrated program. Such effects can be dealt with, if necessary, through the use of neutron damage functions that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects. If additional developments (theoretical or experimental) suggest that the neutron spectra effects might be significant under some conditions, appropriate actions will be taken.

3. The effect of the temperature of irradiation has also been the subject of considerable research. It is well known that radiation damage is less severe at 600°F than at 500°F (the temperature range of concern). The differences in effect on the steel appear to be noticeable and should be taken into account if the irradiation temperature difference is over about 25°F. Enough information is known to permit conservative evaluations of the effect of temperature differences of at least 50°F, and probably even 100°F or more. The differences in the temperature of the surveillance capsules and vessel walls between the B&W power reactors involved in the integrated program are expected to be less than 50°F, and can be conservatively evaluated.

4. The effect of irradiation has also been evaluated by research programs at NRL and other laboratories. Although the consensus of experts on this subject is that there will be no major differences in material property changes by irradiation rates varying over 2 to 3 orders of magnitude, more data from surveillance programs are needed to provide verification. However, the differences in the rates of irradiation of specimen in the integrated program and the limiting material in the walls of the affected vessels will be less than one order of magnitude, therefore, the staff has concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

<u>Conclusion</u>

We have evaluated the adequacy of the proposed integrated, augmented reactor vessel material irradiation program for Oconee 1, 2 and 3 as an alternative to the original program that was interrupted by failure of the associated hardware. We conclude that the proposed program will provide the information required to comply with Appendix G, 10 CFR 50, and that the uncertainties involved in using data obtained from surveillance specimens irradiated in various other B&W power reactors to establish Oconee 1, 2 and 3 vessel operating limitations are small and can be accounted for by imposition of appropriate margins.

Additionally, the proposed integrated, augmented program (with possible minor modification yet to be finalized) should provide more useful information than could have been extracted from the original surveillance program. The proposed program will also give results of the kind required to meet Paragraph V.C of Appendix G, 10 CFR 50.

Until the results of the proposed surveillance program become available, the staffs' predictions of radiation damage in the Oconee vessels will be based on the current revision of Regulatory Guide 1.99. At present, this is Revision 1. Because the chemical analyses of the B&W welds have shown considerable variation, the staff intends to use the Upper Limit lines as the basis for any prediction required at this time.