UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

INTRODUCTION

By letter dated August 17, 1976, as supplemented by letters dated December 20 and 22, 1976, and January 13. 1977, Arkansas Power & Light Company (AP&L) requested that (1) the exemption to Appendix H of Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) granted for Arkansas Nuclear One - Unit No. 1 (ANO-1) by letter of June 11, 1976, be modified to allow indefinite operation of ANO-1 with the remainder of the reactor vessel surveillance capsules to be irradiated at Davis-Besse Unit No. 1 rather than in-situ, and (2) the ANO-1 Technical Specifications be revised to allow the remainder of ANO-1 reactor vessel surveillance capsules to be irradiated at Davis-Besse Unit No. 1. We have determined that several modifications to the AP&L proposal were necessary to meet regulatory requirements. These modifications have been discussed with and agreed to by the AP&L staff.

DISCUSSION AND EVALUATION

The original ANO-1 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 ANO-1 facility on March 19, 1976, to inspect the SSHTs. The inspection revealed that all of the SSHTs had suffered severe damage and that portions of two SSHTs had fallen to the bottom of the reactor vessel. 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 (Cycle 1) were removed from the vessel.

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Since the discovery of the damage to the SSHTs, Babcock & Wilcox Company (B&W), the reactor supplier, has undertaken the design, manufacture and testing of an improved SSHT. SSHTs of this improved design are presently installed in Davis-Besse Unit No. 1 and Crystal River Unit No. 3 and are planned for installation in Three Mile Island Unit No. 2. All three of these plants have reactors supplied by B&W and all are in the process of beginning initial operation within the next few months. In addition, all of these reactors are of the same basic B&W 177 fuel assembly vessel design as ANO-1. The acceptability of the redesigned SSHTs has been demonstrated by a test program reviewed and approved by the staff and performed in conjunction with the Hot Functional Test performed at Davis-Besse Unit No. 1.

Installation of the redesigned SSHTs in the Davis-Besse Unit No. 1 and Crystal River Unit No. 3 reactor vessels did not present any unusual difficulties because it was performed prior to neutron activation of the reactor internals. This will also be the case at Three Mile Island Unit No. 2. 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 ANO-1, they would cause significant radiation to personnel. Based on their experience in removing the SSHTs at Three Mile Island Unit No. 1 and Rancho Seco Unit No. 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 re-installing the SSHTs in ANO-1 and the other irradiated B&W plants.

This program is complex and 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 at operating reactors which suffered damage to SSHTs 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 surveillance program for reactors such as ANO-1 would be 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 which has been fitted with the newly-designed capsule holders on the thermal shield in less time and without radiation exposure of the workmen;

2. An augmented surveillance feature in which more weld metal specimens and some larger fracture mechanics [compact tension (CT)] specimens will be included in the program; and

3. A data-sharing feature in which all available irradiation data for all of the beltline welds of a given reactor vessel will be considered by the licensee or his consultants in predicting the adjusted reference temperature and in making any fracture analyses for that vessel. 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 ANO-1 involves installing the remaining original ANO-1 surveillance capsules (one has been removed and tested) in extra locations provided in the Davis-Besse Unit No. 1 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 ANO-1 reactor vessel 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 Unit No. 3, Three Mile Island Unit No. 2, and Davis-Besse Unit No. 1 on material considered to be essentially identical to the actual welds in the ANO-1 vessel. It is also important to note that still more information relevant to the ANO-1 vessel materials will be obtained from the NRC sponsored Heavy Section Steel Technology (HSST) irradiation programs underway. Details are provided below.

Two weld materials are of primary interest for the ANO-1 vessel, Procedure Qualification (P.Q.) numbers* WF 112 and WF 182-1. These are used in

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*Weld materials 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.

the top and center circumferential welds. The end of life (EOL) fluence for both of these welds is estimated to be 1.1×10^{19} nvt, and both have compositions that are expected to make them relatively sensitive to radiation damage. Weld P.Q. No. WF 18, used for the longitudinal welds, has low copper, making it less sensitive to radiation. Further, the EOL fluence at the azimuthal locations of these longitudinal welds is low (7×10^{18}) so they will not become limiting during the service life. Another shell weld, the lower circumferential, is made of a material that is expected to be radiation sensitive (P.Q. No. SA 1788), 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 is not expected to be limiting.

The orginal ANO-1 surveillance material, WF 193, used the same heat of filler wire as WF 112 but a different batch of flux. Metallurgical considerations suggest that the radiation behavior is affected more by the wire than the flux, thus it is expected to respond to radiation much like WF 112.

The following table shows where samples of these three 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.

WELD	CAPSULE DESIGNATION*	IRRADIATION LOCATIONS	INFORMATION AVAILABLE	SPECIMEN TYPES **
WF 112	OCI-E	Oconee Unit No. 1	1977	Cv (already removed)
	OCI-A	Crystal River Unit No. 3	1985	Cv
	OCI-C	Crystal River Unit No. 3	1989	Cv
WF 182	TMI-2-B	Three Mile Island Unit No. 2	1985	Cv, CT
	TMI-2-D	Three Mile Island Unit No. 2	1995	Cv
	TMI-2-F	Three Mile Island Unit No. 2	1998	Cv, CT
	TE1-F	Davis-Besse Unit No. 1	1979	Cv, CT
	TE1-B	Davis-Besse Unit No. 1	1983	Cv, CT
	TE1-D	Davis-Besse Unit No. 1	1992	Cv, CT
WF 193	AN1-E	Arkansas Unit No. 1	1977	Cv (already removed)
	AN1-A	Davis-Besse Unit No. 1	(to be	Cv
	AN1-C	Davis-Besse Unit No. 1	determined)	Cv

The irradiation schedule and withdrawal dates shown will be modified as initial test results are obtained and evaluated to optimize the information obtained.

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*OCI-E - means capsule E from the Oconee Unit No. 1 reactor
TMI-2-B - means capsule B in the Three Mile Island Unit No. 2 reactor
TE1-F - means capsule F in Toledo Edison's Davis-Besse Unit No. 1 reactor
ANI-E - means capsule E from Arkansas Nuclear One - Unit No. 1 reactor
**Cv - means Charpy V-notch specimen
CT - means Compact Tension specimen

In addition to this integrated program, "research" capsules containing tensile, Charpy V-notch (Cv), and several sizes of CT specimens will be included in the overall B&W power reactor surveillance program. Samples of the weld most likely to be limiting in ANO-1, P.Q. WF 112, will be irradiated in Davis-Besse, and samples of a weld made of the same heat of weld wire as WF 182-1 will be irradiated in the Crystal River Unit No. 3 program. Details of withdrawal schedules will be determined later, and will depend on test results from the other programs.

Research programs being sponsored by the NRC will also provide useful 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 the reference temperature for nil-ductility (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.

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 ANO-1 reactor vessel will be sufficient to provide assurance of safety margins against vessel failure that comply with Appendix G, 10 CFR 50.

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 five years of operation. This Regulatory Guide is based on the NRC staff's analysis of all data available at the time it was written. New data, in particular the results of the augmented integrated surveillance program described above, will be used to periodically update the Regulatory Guide. Predictions of the adjustment of reference temperature and the drop in upper shelf energy are given graphically in R.G. 1.99 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 NRC staff intends to use the Upper Limit lines as the basis for any prediction required at this time.

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*Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", July 1975. Revision 1 is to be published in April 1977.

We have also considered the uncertainties involved in applying radiation effects information obtained in other reactors to the ANO-1 vessel. 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.

1. Calculational methods for estimating the neutron flux at the reactor vessel wall and at 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. As a result, the fast neutron flux and fluence in these locations can generally be calculated to an accuracy of $\pm$ 20%, particularly if some dosimetry checks are available. Dosimeters from the original ANO-1 surveillance program were removed and tested, so the fluence calculations for the vessel can be verified.

   In addition, it should be noted that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence; hence, uncertainties of 20 to 50% in fluence are not highly significant.

   We have also considered the fact that the design of the ANO-1 vessel, internals, and core is nominally identical to that of the other reactors which will be used to obtain radiation effects information.

   These considerations are the basis for our 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 stated to be relatively minor, because they have similar geometry.

   We 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 methods that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects. The neutron spectrum uncertainty will be kept under active scrutiny by the NRC staff and if additional developments (theoretical or experimental) suggest that the effect might be significant under some conditions, appropriate adjustments in reference temperature, drop in upper shelf energy or other suitable parameter can be made.

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 rate has also been evaluated by research programs at the Naval Research Laboratory (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 specimens in the integrated program and the limiting material in the walls of the affected vessels will be less than one order of magnitude. Therefore, we have concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

CONCLUSION

We have evaluated the adequacy of the proposed integrated, augmented reactor vessel material irradiation program for ANO-1 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 ANO-1 vessel operating limitations are small and can be accounted for by imposition of appropriate margins. We also conclude that the associated Technical Specification changes to implement the program are acceptable.

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, our predictions of radiation damage in the B&W power reactors will be based on the current revision of Regulatory Guide 1.99. At present, this is Revision 1.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSIONS

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability of consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: