NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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Mr. Benard C. Rusche	Duke Power Company Charlotte, North Carolina Mr. William O. Parker, Jr.	1/4/77
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DESCRIPTION

Ltr. re our 11/16/76 ltr. and their 12/9/76 ltr....trans the following:

ENCLOSURE

Consists of responses to questions constituting the substance of their proposed integrated reactor vessel material surveillance program at the Crystal River site...

ACKNOWLEDGED

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PLANT NAME: (2-P) (26-P)

Oconee Units 1-2-3

SAFETY	FOR ACTION/INFORMATION	ENVIRO 1/13/77 RJL
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NRC FORM 195 (2-76)

DUKE POWER COMPANY
POWER BUILDING
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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

January 4, 1977

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactor Branch #1

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

Dear Mr. Schwencer:

Your letter of November 16, 1976 pointed out that exemptions from the provisions of 10CFR50, Appendix H had been issued to permit the operation of Oconee Units 1, 2 and 3, respectively for their present fuel cycles with the reactor vessel material specimens removed from their reactor vessels. It was stated that review of a course of action other than reinstallation of specimens in the vessel from which they were removed may require an extended period of time on the part of the staff. Accordingly, our plans for obtaining compliance with 10CFR50, Appendix H were requested, and if other than reinstallation in the original vessel was chosen, additional information was to be provided.

Our letter dated December 9, 1976 stated that it is our intention to continue the irradiation of the Oconee surveillance capsules in a reactor of similar design. To this end, an agreement in principle has been reached with Florida Power Corporation for the irradiation of Oconee capsules in the Crystal River, Unit 3 reactor vessel. It is expected that this agreement will be formalized in the near future. The attached responses to questions constitute the substance of our proposal to NRC to permit this integrated reactor vessel material surveillance program at the Crystal River site. In the attached, host reactor and guest reactors are Crystal River 3 and Oconee, respectively. Formal proposal of this program and the necessary Technical Specifications will be submitted in the near future.

Your letter also requested a description of any additional programs that we plan to implement to satisfy the fracture toughness requirements of Appendix G to 10CFR50. Such a program would involve additional surveillance capsules irradiated at operating reactors and/or test reactor. We

have participated in the B&W User's Group effort toward such a program. At this time, no formal agreement for such a program has been reached. Information on this program will be provided when available.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr. by

MST:ge

RESPONSES TO NRC QUESTIONS ON
REACTOR VESSEL SURVEILLANCE PROGRAM

1. Provide your contingency plans for assuring that your surveillance program will not be jeopardized by an extended outage of any other reactor(s) from which you expect to receive data. What time limits will you place on the those reactor(s) for a given outage and justify these limits.

RESPONSE:

B&W has developed a combined program for irradiating surveillance specimens of welds of interest between operating reactors and test reactors. Such a synergistic program will offer protection against an extended outage of the host reactor should this occur. Redundancy will be incorporated in the combined program by ensuring that most of the representative welds to be irradiated in operating reactors will also be irradiated in the test reactors. The fluence levels in the test reactor programs should be sufficiently high to ensure that the surveillance material stays ahead of the corresponding reactor vessel beltline region. This, in itself, will allow for somewhat other than normal outages at the host reactor. Also, there is redundancy incorporated in the operating reactor program so that if an outage occurs at one host reactor, at least one other host reactor will have representative weld metal in a neutron environment. In summary, the combined surveillance program offers a double redundancy feature for the irradiation of representative weld metal should the host reactor suffer an extended outage.

There is no time limitation on an outage at the host reactor. The operations of this plant will be monitored as discussed in response to question 5 at the frequencies specified in response to question 6. Should it be determined that an extended outage has the potential for allowing the fluence on the guest reactor vessel to approach the fluence on the surveillance capsules at the host reactor, a review of alternative sources of surveillance data will be made, as discussed in response to question 4. The length of time that the host reactor can remain out of service is, of course, a function of the prior service.

2a. Provide your program and schedule for installing the redesigned surveillance capsule holders in your reactor in the event this action becomes necessary.

RESPONSE:

Due to the availability of applicable surveillance data from alternate sources it is not expected that installation of surveillance specimen holder tubes (SSHT) will be necessary. In the remote event it does become necessary. B&W must first complete the development and testing of a substantial amount of required tooling. A tabulation of the required tooling and its current status is given in Table 1.

B&W estimates that 26 months will be required to complete the development and testing of the above tooling. This 26 months will have to be expended before holder tube installation can be initiated. Once the tooling is developed an estimated 3 months will be required to install three surveillance capsule holders on an irradiated plant. This time estimate does not include any contingency for an improperly installed tube or failure of any tooling to perform as planned and tested.

At this time, B&W is not proceeding with continued development of tooling for the installation of surveillance capsule holders on irradiated plants. Once such a requirement is identified, a time span of at least 29 months must transpire before a plant can be outfitted with installed holder tubes.

2b. What is the schedule for withdrawal of your capsules from the host reactor(s)? Relate the schedule to predicted trends in adjusted reference temperature and Charpy upper shelf energy. What arrangements have been made with the owners of the host reactors to assure that this withdrawal schedule will be met.

RESPONSE:

Table 2 lists the withdrawal schedule for the surveillance capsules, as related to the appropriate cycle at the host reactor. Table 3 presents the basis and justification for this withdrawal schedule. Table 3 relates the schedule to the actual and predicted trends in adjusted reference temperature and charpy upper shelf energy of the surveillance weld metal.

Duke Power Company and Florida Power Corporation have agreed in principle on arrangements for the irradiation of the Oconee capsules at Crystal River Unit 3. This agreement commits Florida Power to making every reasonable effort to assist Duke in meeting the capsule withdrawal schedule. As of this date, the agreement has not been formalized, pending Florida Power's receipt of a full power operating license for Crystal River Unit 3.

3. Specify the minimum and maximum radiation lead times for: (a) surveillance specimens relative to the vessel beltline inner surface, and (b) surveillance specimens relative to the 1/4T position in vessel wall, which you will require for guest specimens exposed in the host reactor(s). Justify the values specified.

RESPONSE:

The minimum radiation lead time specified is 2.5 equivalent effective full power years for the surveillance specimen capsules at the host reactor relative to the guest reactor vessel surface location. This limit is based on assuring that surveillance data with adequate lead time will be available to use in reviewing the heatup and cooldown pressurization limits as required by 10CFR50 Appendix H, Section IV. Such data will assure that the requirements of 10CFR50, Appendix G, Section V.B. are met for the future service period.

The minimum lead time relative to the reactor vessel 1/4 thickness location is not specified since it will always be a greater lead time than the surface due to neutron attenuation through the vessel wall. Maximum lead times are not specified, since the withdrawal schedule discussed in response to question 2b will provide the required surveillance data at the proper intervals of service life regardless of the operational status of the guest reactor. Should the guest reactor's service be interrupted for an extended period of time with continued operation of the host reactor, the effect would be the ability to verify vessel material properties for a longer period of service than would otherwise be possible. Thus no limits are required on maximum lead time.

The 2.5 EFPY lead time specified includes consideration of the time required to develop alternative means of obtaining the required surveillance data should the host reactor experience an extended outage. The longest lead time alternative is the ultimate installation of surveillance specimen holder tubes (SSHT) at an irradiated plant and the expected 37 months (2.5 EFPY at .8 capacity factor) provides ample time for the necessary tooling development and installation of SSHT's should this be necessary.

Implementation of the 2.5 EFPY lead time is not necessary until the beginning of Oconee 1 Cycle 7 or Oconee 2 and 3 Cycle 5 or later since the heatup and cooldown pressurization limits are being conservatively reviewed for adequacy based on presently available surveillance data and conservative estimates on materials similar to those of the Oconee reactors vessel exposed to this range of fluence. This review is presently planned to be completed in early 1977 as part of the evaluation of the surveillance capsules which were tested in 1976.

4. Indicate the corrective action to be undertaken at the guest reactor if the limits specified in response to Question 3 above cannot be met. If the corrective action does not involve reactor shutdown, justify the proposed alternative.

RESPONSE:

Since there are several alternatives, the corrective action will not involve reactor shutdown. As discussed in the answer to Question 1, B&W has developed a synergistic surveillance program in which several welds and base metals will be irradiated in three operating reactors (Davis Besse 1, Crystal River 3, and Three Mile Island 2). In addition, data for several of the same welds will be obtained in at least two test reactor irradiation programs. The welds to be irradiated in the test reactor programs are described in the responses to Questions 11 thru 16. The synergistic program assures that applicable data for the 177 FA, B&W design reactor vessels, will be available through the design service life of the vessels.

In the event that the host reactor has an extended outage which is of sufficient duration to endanger the timeliness of the data availability, several possibilities exist that would minimize the impact of such an outage. Such possibilities or alternatives are:

1. An evaluation of the applicability of the available data (from that reactor or other reactors including test reactors) to the guest reactor could be made. Such evaluation may indicate the guest reactor capsules do not need to be irradiated within the expected time of the host reactor outage.

2. The capsules which will generate applicable data for the guest reactor can be removed from the host reactor that is shutdown and inserted into another host reactor that is in operation.

3. The pressure-temperature limit curves of the guest reactor could be developed with material properties conservatively assumed until applicable data is available.

The best alternative can only be chosen at the time at which the extended outage occurs, since all the above options require evaluation of the data which is or will be available in a timely manner. The monitoring program described in response to Questions 5 and 6 provides assurance that the potential for untimely data will be identified in time so that none of the alternatives, including eventual installation of SSHT's on an irradiated plant, is precluded.

5. Describe how the operating staff of the guest reactor will keep informed of the exposure status of the guest specimens at the host reactor(s) relative to the limits specified in response to Question 3 above.

RESPONSE:

A procedure has been developed to determine fluence lead time based on conservatively estimated fluence values as a function of effective full power years (EFPY) for the guest and host reactors. The lead time is defined as the equivalent EFPY of predicted neutron fluence received by the guest reactor at the reactor vessel surface subtracted from the equivalent EFPY of predicted neutron fluence received by the reactor vessel surveillance capsules at the host reactor. The procedure also considers the initial irradiation received by the capsules during their original exposure in the guest reactor prior to SSHT removal.

The basic equation for determining the lead time (in EFPY) is:

$$\text{Lead Time} = T_{E,i} - T_E$$

where: T_E = Actual cumulative EFPY on the guest reactor based on licensed core power

$T_{E,i}$ = Calculated equivalent EFPY on the capsules in the host reactor

$T_{E,i}$ is calculated from the following equation:

$$T_{E,i} = C_{2,i} + C_{3,j}\, T_H$$

where: $C_{2,i}$ = Constant dependent on initial equivalent fluence on the capsules

$C_{3,j}$ = Equivalent fluence factor from the host reactor capsule location to the guest reactor vessel wall

T_H = EFPY of the host reactor, cumulative

6. Submit amended proposed Technical Specifications that reflect the appropriate portions of your responses to Questions 2, 3, 4 and 5 above.

RESPONSE:

The following is an example of the Technical Specifications which will be formally submitted in the near future.

4.2.13

Starting with initial operation of Oconee 1, Cycle 4 or Oconee 2 and 3, Cycle 3, whichever occurs first, the reactor vessel surveillance specimen capsule lead time shall be determined at the frequencies specified in Table 4.2-1. Subsequent to Oconee 1, Cycle 7 or Oconee 2 and 3, Cycle 5 operation, whichever occurs first, if the lead time is less than 2.5 EFPY, a report describing the means of providing the necessary reactor vessel surveillance data shall be submitted for NRC review within 150 days of calculating a lead time of < 2.5.

Table 4.2-1

SURVEILLANCE SPECIMEN CAPSULE IRRADIATION COMPARISON FREQUENCY

Capsule Lead (EFPY)	Comparison Frequency
< 3.5	Quarterly
3.5 < 4.5	Semi-annual
> 4.5	Annual

The equivalent effective full power years (EFPY) of predicted neutron fluence (E > 1MEV) received by the guest reactor vessel at the surface location shall be subtracted from the equivalent EFPY of predicted neutron fluence (E > 1 MEV) received by the reactor vessel surveillance specimen capsules at the host reactor to determine the capsule lead factor. The comparison frequency is to be determined by the lead factor calculated at the last scheduled comparison.

Bases

To assure the availability of adequate surveillance data for the Oconee reactor vessels, a program has been developed to monitor the irradiation of the surveillance specimen capsules at the host reactor, and compare this to the irradiation of the guest reactor vessel. Fluence estimates which are conservative in the appropriate direction are used for this comparison. The frequency of monitoring varies depending on the known neutron fluence lead factor between the capsules and the reactor vessel. This provides ample time for anticipating problems and initiating corrective action should operation of the host reactor be seriously delayed. The requirement that the lead factor be 2.5 EFPY by the end of Oconee 1, Cycle 6 or Oconee 2 and 3 Cycle 4, or corrective action be developed provides assurance that surveillance data will be available in a timely manner to allow revisions to Technical Specification 3.1.2.3. The lead factor of 2.5 EFPY is based on a .8 capacity factor and thus provides over 3 calendar years for consideration and implementation of all alternatives. The requirement of a factor of 2.5 EFPY lead time need not be implemented prior to operation for the fuel cycles indicated above since Technical Specification 3.1.2.3 is being reviewed for adequacy and will be updated for a period of approximately 10 EFPY total operation. This is based on preliminary results of recent surveillance capsule test results, which justify a period of operation in excess of the fuel cycles indicated above.

7. Provide a comprehensive tabulation for the guest reactor and each host reactor, of the values of all parameters, including construction and operating characteristics, that may affect the fracture toughness of the reactor vessel material as it is irradiated. Discuss how all differences in these parameters are accommodated in the integrated surveillance program.

RESPONSE:

The reactor parameters which could possibly affect the material properties as the vessel is irradiated are (1) the neutron flux energy spectrum, (2) the irradiation rate, (3) the irradiation temperature and (4) the material type and initial properties. Each of these is addressed below.

Energy Spectrum - As discussed in the response to Questions 8, 9 and 10, the relative neutron energy spectrum is primarily a function of the geometry and materials of the reactor internals components. As shown in Table 4, the dimensions and materials of both the host and guest reactors are essentially identical. Thus, there is no difference to be accommodated.

Irradiation Rate - Any significant difference in dose rate obtained at the guest and host reactors would be due to the variations in power level and power distribution. Since the licensed power levels are comparable, the only difference is the variation in load swings as the plant maneuvers. When time averaged over anticipated fuel cycles the variation in power level and power distribution due to maneuvers is expected to be comparable between plants. The comparability of reactor vessel surveillance results from a number of plants presently available, supports this conclusion.

Irradiation Temperature - There are two differences in irradiation temperature considered. The guest reactor vessel beltline inner surface and the surveillance specimens in the host reactor are exposed to reactor coolant at essentially inlet conditions. The temperature distribution in the surveillance specimens and capsules is controlled primarily by the temperature of the reactor coolant. This is due to the good heat transfer characteristics of the specimen/capsule configuration. Thus variation in reactor coolant inlet temperatures due both to design difference and the variation as the plant is maneuvered must be considered. The variation due to design differences between the host and guest reactors is insignificant as shown on Table 4. Between partial ($\sim$15%) and full load conditions, the inlet temperature will vary by about 20°F as an inverse function of power level. Figure 4-9 in the FSAR shows this variation. The duration of this variation due to maneuvering is expected to be comparable between plants over time. This is supported by the comparability of reactor vessel surveillance results presently available from a number of plants. In any case, the inlet condition temperatures are considered too low to cause significant annealing. The inlet temperature will also vary about 40°F between the hot zero power condition and partial load. This variation is a direct function of power level (0-15%) and again is not significant due to the low temperature and the expected comparability in duration over the long term.

Material Type and Initial Properties - Both the host and guest reactors are constructed of similar materials as discussed in conjunction with the neutron spectrum consideration. Thus, there is no difference to be accommodated.

8. Describe analytical techniques that you plan to use to estimate the fluence expected at the various welds of the beltline of your vessel. How much uncertainty do you expect there to be in the fluence estimates?

RESPONSE:

Energy dependent neutron fluxes are determined by a discrete ordinate solution of the Boltzmann transport equation. Specifically, ANISN, a one-dimensional code, and DOT, a two-dimensional code, are used to calculate the flux at the detector position. In both codes, the system is modeled radially from the core out to the air gap outside the pressure vessel. The model includes the core with a time averaged radial power distribution, core liner, barrel, thermal shield, pressure vessel, and water regions. Inclusion of the internal components is necessary to account for the distortions of the required energy spectrum by attenuation in these components. The ANISN code uses the CASK 22-group neutron cross section set with an S_6 order of angular quadrature and a P_3 expansion of the scattering matrix. The problem is run along a radius across the core flats. Azimuthal variations are obtained with a DOT r-theta calculation that models one-eighth of a plan-view of the core (at the core midplane) and includes a pin by pin, plant specific time averaged power distribution. The DOT calculation uses S_6 quadrature and a P_1 cross section set derived from CASK.

Fluxes calculated with this DOT model must be adjusted to account for lack of P_3 cross section detail in calculations of anisotropic scattering, a perturbation caused by the presence of the capsule, and the axial power distribution. The first two items are both energy and radial-location dependent whereas the latter is axial location dependent. A P_3/P_1 correction factor is obtained by comparing two ANISN 1-D model calculations in which only the order of scatterin was varied. The capsule perturbation factor is obtained from a comparison of two DOT x-y model calculations, one with a capsule explicitly modeled - SS304 cladding, Al filler region, and carbon steel specimens--and the other with water in those regions. The effect of axial power distribution is determined from plant specific burnup calculations as a function of axial location for the outer rows of fuel assemblies. The net result from these parameter studies is a flux adjustment factor K which is applicable to the appropriate dosimeters in the 177-FA surveillance programs.

The calculation described above provides the neutron flux as a function of energy at the dosimeter position. These calculated data are used in the following equations to obtain the calculated activities used for comparison with the experimental values. The basic equation for the activity D (in μCi/gm) is given as follows:

$$D_i = \frac{CN}{A_i \; 3.7 \times 10^4} \; f_i \sum_E \sigma_n(E)\phi(E) \sum_{j=1}^{M} F_j \, (1-e)^{-\lambda_i \tau_j} e^{-\lambda_i (T-\tau_j)}$$

where

C = normalizing constant, ratio of measured to calculated flux

N = Avogadro's number

A_i = atomic weight of target material i

f_i = either weight fraction of target isotope in nth material of fission yield of desired isotope

$\sigma_n(E)$ = group-averaged cross sections for material n listed in Table D-3

$\phi(E)$ = group-averaged fluxes calculated by DOT analysis

F_j = fraction of full power during jth time interval, t_j

λ_i = decay constant of ith material

t_j - interval of power history

T = sum of total irradiation time, i.e., residual time in reactor, and wait time between reactor shutdown and counting

τ_j = cumulative time from reactor startup to end of jth time period, i.e.,

$$\tau_j = \sum_{k=1}^{j} t_k \, .$$

The normalizing constant C can be obtained by equating the right side of the above equation to the measured activity. With C specified, the neutron fluence greater than 1 Mev can be calculated from

$$\phi(E > 1.0 \text{ MEV}) = C \sum_{E=1}^{15 \text{ Mev}} \phi(E) \sum_{j=1}^{M} F_j t_j$$

where M is the number of irradiation time intervals; the other values are defined above.

The analytical model described above, for calculating fast fluence at the surveillance capsule includes the pressure vessel region. Thus each calculation produces fluence data at the weld position as well as the capsule location. Since analytical results currently being documented compare within ± 15% to dosimeter measurements from surveillance capsules from 5 reactors to date, calculated data at the nearby weld position should have similar reliability. Dosimeter data comparisons from surveillance capsules irradiated at the host reactor will provide further comparisons

with the analytical model. Because of the similarity of the host and guest reactors, these comparisons will also be applicable to the pressure vessel fluence calculation for the guest reactor since it uses the same analytical model.

B&W intends to document the uncertainty based on the contributing factors in both the calculation and the measurements from the present capsule evaluations (the value should be on the order of ± 30%). This documentation will be available following completion of the surveillance capsule results and submittal to NRC in the form of a Topical Report is expected by June 1977.

9. Describe any dosimetry checks that you plan to make on the analytical results.

RESPONSE:

Dosimeter measurements from Oconee 1 (Cycles 1 and 2), Oconee II (Cycle 1), Oconee III (Cycle 1), TMI-1 (Cycle 1) and ANO-1 (Cycle 1) have been compared to the analytical model. A nominal difference of ± 15% was noted in the fast flux (E>1 MEV). Multiple dosimeters in surveillance capsules will be in the host reactors and also in subsequent B&W plants to startup in the 1980's. When each capsule is removed dosimeter activities will be measured and then compared to the plant specific analytical result. This will provide data for further vertification comparisons with the analytical technique which will be used for plant specific fluence calculations at both the host and guest reactors. No check is considered necessary for calculated data at the weld locations as noted in response to Question 7.

10. What differences in neutron energy spectra and dose rate do you predict for your reactor beltline and your surveillance specimens, wherever they are to be irradiated? Describe the corrections, if any, that will be made to the predicted radiation damage at your beltline welds as a result of these differences. Possible corrections include differences in specimen irradiation temperatures, differences in neutron spectra arising from differences in reactor geometry or a different type of fuel (e.g., mixed oxides), and differences in dose rate if some test reactor data are used.

RESPONSE:

For the same fuel type (e.g., low enriched uranium), relative neutron energy spectrum is a function of only the internals components (geometry and materials). The internal components design is the same for both guest and host reactors as discussed in response to Question 7. Thus the relative energy spectrum at the same spatial location should not vary between reactors. Therefore, dose rate will vary directly as the fast flux. The analytical model is a multigroup calculation with the same internals arrangement using plant specific core parameters as discussed in response to Question 7. Consequently no correction is required between plants since the significant variables are already accounted for in the calculation. The use of mixed oxide fuel would harden the spectrum somewhat but any effect on dose rates should be within the analysis uncertainty limits. Possible corrections in using data from test reactors will depend on the design of the test reactor program, which is not final.

Since impact, tensile and fracture data on many of the same materials, will be obtained both from test reactors and the surveillance programs, a basis for comparison will be available. Such comparison will determine if correction would be needed.

11. Identify the heats of weld wire and flux used in all beltline welds, and give specific locations where each is used.

RESPONSE:

The heats of weld wire and flux used in all beltline region welds, including the surveillance weld, and their specific locations are given in Tables 5a, 5b, and 5c for Oconee I, Oconee II and Oconee III, respectively.

12. State which weld or welds is expected to be controlling with regard to radiation damage and why, i.e., give expected neutron flux, initial RT_{NDT}, Charpy upper shelf energy, and chemical composition for the controlling welds.

RESPONSE:

For Oconee I - Table 5a also lists the unirradiated RT_{NDT} and Charpy upper shelf energy (Cv-USE), the weight percent of the pertinent elements, the expected end of service neutron fluence at the 1/4T vessel wall location, the predicted shift and adjusted RT_{NDT}, and the predicted drop and adjusted Cv-USE. As shown in Table 5a weld SA 1585 has the highest adjusted RT_{NDT} and the lowest adjusted Cv-USE of all the beltline region welds. However, welds WF 25, and SA 1229 also have the potential of being the controlling weld because their predicted irradiated properties are similar to those predicted for weld SA 1585. The surveillance weld, WF 112, is considered to be representative of the controlling welds. The predicted irradiated properties for the surveillance weld are similar to those predicted for WF 25, SA 1229 and SA 1585 at the same fluence value. Note that the unirradiated properties of WF 25 and WF 112 were determined by testing and those for SA 1229 and SA 1585 are estimated.

For Oconee II - Table 5b also lists the unirradiated RT_{NDT} and Charpy upper shelf energy (Cv-USE), the weight percent of the pertinent elements, the expected end of service neutron fluence at the 1/4T vessel wall location, the predicted shift and adjusted RT_{NDT}, and the predicted drop and adjusted Cv-USE. As shown in Table 5b weld WF 25 has the highest adjusted RT_{NDT} and weld WF 154 has the lowest adjusted Cv-USE of all the beltline region welds. The other beltline region weld, WF 112, is subjected to a very low neutron fluence and its properties are expected to be the same throughout the service life of the plant. Welds WF 25 and WF 154 both have the potential of being the controlling weld. The surveillance weld, WF 209-1, is considered to be representative of the controlling welds. The predicted irradiated properties for the surveillance weld are similar to those predicted for WF 25, and WF 154 at the same fluence value. Note that the unirradiated properties of WF 25, WF 112, and WF 209-1 were determined by testing and those for WF 154 are estimated.

For Oconee III - Table 5c also lists the unirradiated RT_{NDT} and Charpy upper shelf energy (Cv-USE), the weight percent of the pertinent elements, the expected end of service neutron fluence at the 1/4T vessel wall location, the predicted drop and adjusted Cv-USE. As shown in Table 5c welds WF 67 and WF 70 both have the same predicted irradiated properties. Both welds have the highest adjusted RT_{NDT} and the lowest adjusted Cv-USE of all the beltline region welds. Weld WF 200 also has the potential of being the controlling weld because its copper content is only 0.01% lower than those determined for WF 67 and WF 70. The unirradiated properties of all the beltline region welds are estimated. The surveillance weld. WF 209-1, is considered to be representative of the controlling weld. The predicted adjusted RT_{NDT} for the surveillance weld is 45°F higher than that predicted for WF 67 and WF 70, however, the adjusted Cv-USE is about the same. Even though the adjusted RT_{NDT} for the surveillance weld is higher than for the controlling weld, the surveillance weld is considered representative because welds WF 70 and WF 209-1 have the same weld wire heat number.

13. Which welds are represented in the surveillance capsules irradiated in your reactor?

RESPONSE:

See the response to Question 12.

14. Which welds, if any, are represented in surveillance programs for other reactors?

RESPONSE:

Table 6 lists all the welds that are considered representative which will be irradiated as part of the surveillance program of this and other 177FA B&W design power plants. The welds of Table 6 are considered representative of the beltline region welds.

15. List any test reactor programs on radiation damage in which your weld metals are represented.

RESPONSE:

Presently there are two test reactor programs in which representative welds will be studied. These programs are:

1. HSST Irradiation Studies Program.

2. NRC-NRL Implace Annealing Studies Program.

Data from these programs is, of course, readily available to NRC.

16. List any other test reactor and surveillance programs in which welds that are expected to be in the same category as yours from the standpoint of radiation sensitivity are represented, which you intend to utilize.

RESPONSE:

Other than the programs outlined in response to Question 14 and 15 B&W

is investigating the possibilities of irradiating similar weld metals in a EPRI program to be initiated prior to mid-1977.

TABLE 1 STATUS OF REQUIRED SSHT IRRADIATED PLANT INSTALLATION TOOLING

	Tool	Status	Comments
1.	Boring Mill	Complete with Backup	
2.	Pintle Removal Tool	Complete, no Backup	No backup Necessary
3.	Drill and Tapper	a) Basic tool 98% complete. b) Backup drills and taps must be sealed water-tight and tested. c) Drilling and tapping at other than pintle locations has not been developed.	
4.	Thread Inspection Tool	Concept only	
5.	Spot Face Tool	Basic tool 20% complete	
6.	Spot Face Inspection Tool	Concept only	
7.	S.S.H.T. Installation Tool	Concept only	
8.	Verification of Bracket Contact Inspection Tool	Concept only	
9.	Crimping Tool	Concept only	
10.	Free Path Inspection Tool	Complete, no Backup	

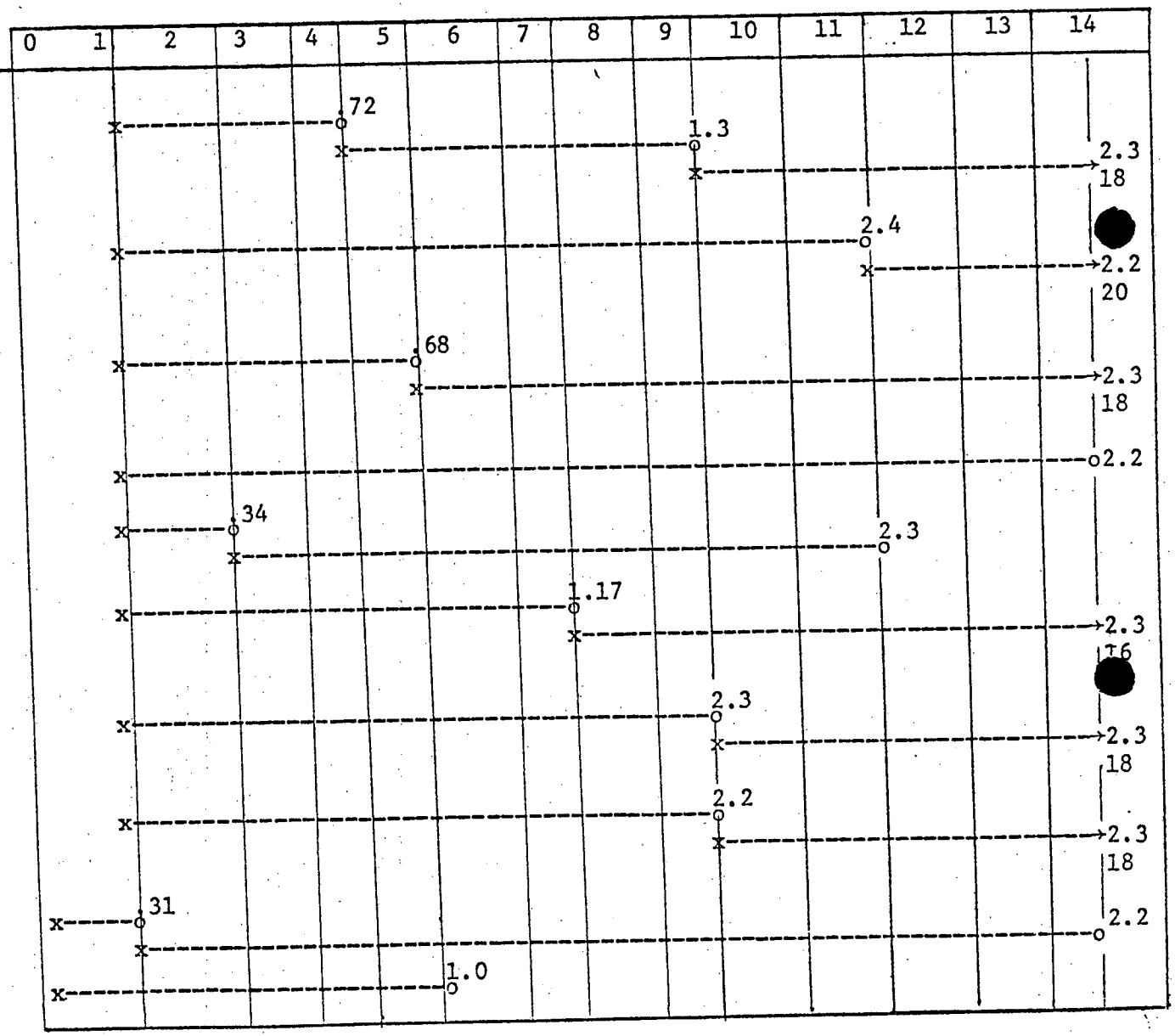
TABLE 2

INSERT AND WITHDRAWAL SCHEDULE OF

INTEGRATED PROGRAM AT CRYSTAL RIVER 3

CYCLES

Holder Tube	Lead Factor at 1/4T	Location	Capsule
WZ	9.6	Upper	CR3-L1 OCII-B OCII-D
		Lower	CR3-L2 OCIII-F
ZY	6.9	Upper	CR3-C OCIII-E
		Lower	CR3-A
YX	9.6	Upper	OCII-A OCI-C
		Lower	OCI-A OCI-B
YX	9.6	Upper	OCII-E OCII-F
		Lower	OCIII-D OCI-D
XW	6.9	Upper	CR3-B CR3-E
		Lower	CR3-D



(cont'd on page 2)

INSERT AND WITHDRAWAL SCHEDULE OF

INTEGRATED PROGRAM AT CRYSTAL RIVER 3

CYCLES

Holder Tube	Lead Factor at 1/4T	Location	Capsule	0	1	2	3	4	5	6	7	8	9	10	11	12	13	14
WX	9.6	Upper	OCIII-B			x-----	o .31											
			OCIII-C			x	----------	----------	----------	----------	o 1.3							
		Lower	CR-3-F			x	----------	----------	----------	----------	----------	----------	----------	----------	o 2.16			

The assumed EFPD per cycle are 450 days for the first cycle and 250 days for the others. The values to the right of o(Identification of Withdrawal) is the predicted accumulated neutron fluence x 10^{19} ($n/cm^2 E>1Mev$) at the capsule location.

x - Capsule Insertion o - Capsule Withdrawal

Table 3a

SCHEDULE FOR WITHDRAWAL OF OCONEE I's REACTOR VESSEL SURVEILLANCE CAPSULES FROM THE CRYSTAL RIVER UNIT 3

Capsule	Time of Withdrawal(1)	Approximate Neutron Fluence to be Accumulated by Capsule ($E > 1$ Mev, n/cm^2)	Predicted Impact Properties of Surveillance Weld Metal	
			RT_{NDT} (F)	Cv-USE (Ft-Lbs)
Unirradiated	-	0	0	65
OCI-F	Has been withdrawn for testing	8.3×10^{17} (2a)	No weld metal	No weld metal
OCI-E	Has been withdrawn for testing	1.5×10^{18} (2a)	120	56
OCI-A	To be withdrawn at the time when the capsule's accumulated neutron fluence ($E > 1$ Mev) correspond to that at 1/4 of ONS-1 reactor vessel wall location at approximately the end of vessel's design service life.	1.2×10^{19} (2b)	220	40
OCI-C	To be withdrawn at the time when the capsule's accumulated neutron fluence ($E > 1$ Mev) corresponds to that of ONS-1 reactor vessel inner wall location at approximately the end of vessel's design service life	2.2×10^{19} (2b)	280	38
OCI-B	Standby	$>2.2 \times 10^{19}$ (2b)	No weld metal	No weld metal
OCI-D	Standby	$>2.2 \times 10^{19}$ (2b)	No weld metal	No weld metal

(1) Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdown of Crystal River 3 most closely approaching the above withdrawal schedule. The schedule may also be modified, if necessary, after the evaluation of each capsule.

(2a) Measured value using dosimeter data.

(2b) Predicted neutron fluence values for ONS-1's vessel. They are measured values extrapolated based on predicted power distribution leakage flux, and fuel handling procedures. Values contain a 1.2 safety factor.

Table 3b

SCHEDULE FOR WITHDRAWAL OF OCONEE II's REACTOR VESSEL SURVEILLANCE CAPSULES FROM THE CRYSTAL RIVER UNIT 3

Capsule	Time of Withdrawal(1)	Approximate Neutron Fluence to be Accumulated by Capsule ($E > 1$ Mev, n/cm^2)	Predicted Impact Properties of Surveillance Weld Metal	
			RT_{NDT} (F)	Cv-USE (Ft-Lbs)
Unirradiated	-	0	+10	68
OCII-C	Has been withdrawn for testing	9.4×10^{17} (2a)	100	54
OCII-A	Following the 2nd cycle at Crystal River 3	3.1×10^{18} (2b)	240	44
OCII-B	To be withdrawn at the time when the capsule's accumulated neutron fluence ($E > 1$ Mev) correspond to that at 1/4 of ONS-2 reactor vessel wall location at approximately the end of vessel's design service life.	1.2×10^{19} (2c)	No weld metal	No weld metal
OCII-E	To be withdrawn at the time when the capsule's accumulated neutron fluence ($E > 1$ Mev) corresponds to that of ONS-2 reactor vessel inner wall location at approximately the end of vessel's design service life.	2.2×10^{19} (2c)	340	36
OCII-D	Standby	$>2.2 \times 10^{19}$ (2c)	No weld metal	No weld metal
OCII-F	Standby	$>2.2 \times 10^{19}$ (2c)	No weld metal	No weld metal

(1) Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdown of Crystal River-3 most closely approaching the above withdrawal schedule. The schedule may also be modified, if necessary, after the evaluation of each capsule.

(2a) Measured value using dosimeter data.

(2b) Predicted neutron fluence value for the capsule in the identified location of Table 2. The assumption made on predicting the fluence value are given in Table 2.

(2c) Predicted neutron fluence values for ONS-2's vessel. They are measured values extrapolated based on predicted power distribution leakage flux, and fuel handling procedures. Values contain 1.2 safety factor.

Table 3c

SCHEDULE FOR WITHDRAWAL OF OCONEE III's REACTOR VESSEL SURVEILLANCE CAPSULES FROM THE CRYSTAL RIVER UNIT 3

Capsule	Time of Withdrawal (1)	Approximate Neutron Fluence to be Accumulated by Capsule ($E > 1$ Mev, n/cm^2)	Predicted Impact Properties of Surveillance Weld Metal	
			RTNDT (F)	Cv-USE (Ft-Lbs)
Unirradiated	-	0	60	66
OCIII-A	Has been withdrawn for testing	7.4×10^{17} (2a)	100	56
OCIII-B	Following the 2nd cycle at Crystal River 3	3.1×10^{18} (2b)	240	44
OCIII-C	To be withdrawn at the time when the capsule's accumulated neutron fluence ($E > 1$ Mev) correspond to that at 1/4 of ONS-3 reactor vessel wall location at approximately the end of vessel's design service life.	1.2×10^{19} (2c)	340	38
OCIII-D	To be withdrawn at the time when the capsule's accumulated neutron fluence ($E > 1$ Mev) corresponds to that of ONS-3 reactor vessel inner wall location at approximately the end of vessel's design service life.	2.1×10^{19} (2c)	380	34
OCIII-E	Standby	$>2.1 \times 10^{19}$ (2c)	380	34
OCIII-F	Standby	$>2.1 \times 10^{19}$ (2c)	380	34

(1) Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdown of Crystal River 3 most closely approaching the above withdrawal schedule. The schedule may also be modified, if necessary, after the evaluation of each capsule.

(2a) Preliminary measured value using dosimeter data.

(2b) Predicted neutron fluence value for the capsule in the identified location of Table 2. The assumption made on predicting the fluence value are given in Table 2.

(2c) Predicted neutron fluence values for ONS-3's vessel. They are measured values extrapolated based on predicted power distribution leakage flux, and fuel handling procedures. Values contain a 1.2 safety factor.

Table 4

Comparison of Oconee -1, -2, -3, and Crystal River-3

Parameter	Oconee-1	Oconee-2	Oconee-3	Crystal River-3
Design Heat Output (Core), MWt	2568	2568	2568	2452
Design Overpower, % Design Power	112	112	112	114*
System Pressure, Nominal, psia	2200	2200	2200	2200
Coolant Flow Rate, lb/hr x 10^{-6}/GPM	131.3/ 352,000	131.3/ 352,000	131.3/ 352,000	131.3/ 352,000
Coolant Temperatures (°F)				
Nominal Inlet	554	554	554	555
Avg. Rise in Vessel	50	50	50	48
Avg. in Vessel	579	579	579	579
Fuel Assemblies, No.	177	177	177	177
Fuel Assemblies, Type	MKB(15x15)	MKB(15x15)	MKB(15x15)	MKB(15x15)
Core Barrel, ID/OD, in.	141/145	141/145	141/145	141/145
Thermal Shield, ID/OD, in.	147/151	147/151	147/151	147/151
Core Structural Characteristics				
Core Diameter, in. (equivalent)	128.9	128.9	128.9	128.9
Core Height, in. (active fuel)	144	144	144	144
Reflector Thicknesses and Composition				
Top (Water plus Steel), in.	12	12	12	12
Bottom (Water plus Steel), in.	12	12	12	12
Side (Water plus Steel), in.	18	18	18	18

Table 4 (Cont'd.)

Comparison of Oconee -1, -2, -3, and Crystal River-3

Parameter	Oconee-1	Oconee-2	Oconee-3	Crystal River-3
Reactor Vessel Design Parameters				
Principal Material	SA-302, Grade B, Class 1, as-modified by Code Case 1339	SA-508, Grade B, Class 2	SA-508, Grade B, Class 2	SA-533, Grade B, Class 1
Design Pressure, psig	2500	2500	2500	2500
Design Temperature, °F	650	650	650	650
ID of Shell, in.	171	171	171	171
OD Across Nozzles, in.	249	249	249	249
Overall Height of Vessel and Closure Head (over Cld and Inst. Nozzles), ft/in.	40-8-3/4	40-8-3/4	40-8-3/4	40-8-7/8
Reactor Internals, Principal Material	304SS	304SS	304SS	304SS

Table 5a

WELD METAL INFORMATION AND DATA

(Oconee I)

Weld Metal Ident.	Weld Metal Wire	Weld Metal Flux	Location in reactor vessel	Unirradiated Impact Data, Transverse RT_{NDT}	Unirradiated Impact Data, Transverse USE	Chemistry Composition % Cu	Chemistry Composition % P	Chemistry Composition % S	Chemistry Composition % Ni	1/4T EOL Neut. Fluence E > 1 Mev n/cm^2 (1)	Shift in RT_{NDT} ΔF	Adjusted RT_{NDT} F	USE Reduction %	Adjusted USE Ft-lbs
WF9	72445	8632	C3	(20)	(66)	.17	.015	.012	.60	$<1.5 \times 10^{17}$	< 20	< 40	<10	>59
WF25	299L44	8650	C1	-15	81	.34	.015	.013	.71	1.2×10^{19}	290	275	44	45
SA1073	1P0962	8445	L1	(20)	(66)	.21	.025	.017	.64	7.6×10^{18}	198	218	33	44
SA1135	61782	8457	C0	(20)	(66)	.17	.015	.013	.50	2.6×10^{18}	115	135	23	51
SA1229	71249	8492	C1	(20)	(66)	.20	.021	.012	.57	1.2×10^{19}	246	266	36	42
SA1430	8T1762	8553	L3	(20)	(66)	.16	.017	.015	.60	1.0×10^{19}	165	185	30	46
SA1493	8T1762	8578	L2	(20)	(66)	.22	.017	.010	.43	7.5×10^{18}	195	215	34	44
SA1585	72445	8597	C2	(20)	(66)	.25	.016	.011	.51	1.2×10^{19}	274	294	40	40
WF112	406L44	8688	SW	0	65	.22	.024	.006	.58	1.2×10^{19}	285	285	37	41

(1) Measured values extrapolated based on predicted power distribution flux leakage, and fuel handling procedures. Values contain a 1.2 safety factor.

NOTE: L1 = Upper longitudinal weld
L2 = Middle longitudinal weld
L3 = Lower longitudinal weld
C0 = Higher-upper circumferential weld
C1 = Upper circum. weld
C2 = Middle circum. weld
C3 = Lower circum. weld
SW = Surveillance weld
( )= Estimated per BAW-10046P

Table 5b

WELD METAL INFORMATION AND DATA

(Oconee II)

Weld Metal			Location in reactor vessel	Unirradiated Impact Data, Transverse		Chemistry Composition %				1/4T EOL Neut. Fluence E > 1 Mev	Shift in RT_{NDT}	Adjusted RT_{NDT}	USE Reduction	Adjusted USE
Ident.	Wire	Flux		RT_{NDT}	USE	Cu	P	S	Ni	n/cm^2 (1)	ΔF	F	%	Ft-lbs
WF25	299L44	8650	C2	-15	81	.34	.015	.013	.71	1.2×10^{19}	290	275	44	45
WF112	406L44	8688	C3	0	65	.22	.024	.006	.58	$<1.5 \times 10^{17}$	< 32	< 32	<10	58
WF154	406L44	8720	C1	(20)	(66)	.20	.015	.021	.59	1.2×10^{19}	214	234	35	41
WF209-1	72105	8773	SW	10	68	.34	.013	.010	.48	1.2×10^{19}	290	300	44	38

(1) Measured values extrapolated based on predicted power distribution flux leakage, and fuel handling procedures. Values contain a 1.2 safety factor.

NOTE: C1 = Upper circum. weld
C2 = Middle circum. weld
C3 = Lower circum. weld
SW = Surveillance weld
( ) = Estimated per BAW-10046P

Table 5c

WELD METAL INFORMATION AND DATA

(Oconee III)

Weld Metal Ident.	Weld Metal Wire	Flux	Location in reactor vessel	Unirradiated Impact Data, Transverse RT_{NDT}	USE	Chemistry Composition % Cu	P	S	Ni	1/4T EOL Neut. Fluence E > 1 Mev n/cm^2 (1)	Shift in RT_{NDT} ΔF	Adjusted RT_{NDT} F	USE Reduction %	Adjusted USE Ft-lbs
WF67	72442	8669	C2	(20)	(66)	.27	.014	.017	.57	1.2×10^{19}	285	305	42	38
WF70	72105	8669	C2	(20)	(66)	.27	.014	.011	.46	1.2×10^{19}	285	305	42	38
WF169-1	8T1554	8754	C3	(20)	(66)	.106	.014	.013	.59	$<1.5 \times 10^{17}$	–	20	–	66
WF200	80IT44	8773	C1	(20)	(66)	.26	.010	.015	.64	1.2×10^{19}	252	270	41	39
WF209-1	72105	8773	SW	60	66	.34	.013	.010	.48	1.2×10^{19}	290	350	44	37

(1) Measured values extrapolated based on predicted power distribution flux leakage, and fuel handling procedures. Values contain a 1.2 safety factor.

NOTE: C1 = Upper circum. weld
C2 = Middle circum. weld
C3 = Lower circum. weld
SW = Surveillance weld

TABLE 6

MATERIAL PROPERTIES OF REPRESENTATIVE WELDS
TO BE IRRADIATED IN SURVEILLANCE PROGRAMS
OF 177 F.A. B&W DESIGN REACTOR VESSELS

Weld Designation	Cu	P	C_v USE(Ft-lbs)	RT_{NDT} (°F)
W1	.40	.020	67	+65
W2	.22	.024	65	0
W3	.24	.016	78	+10
W4	.36	.011	74	-20
W5	.35	.015	72	+10
W6	.24	.022	70	+10
W7	.34	.015	81.3	+ 9
WF25	.29	.019	81	+ 9
WF112	.22	.024	65	0
WF182-1	.18	.014	83	+15
WF193	.19	.016	66	+15
WF209-1	.30	.020	66	+43