Docket Nos. 50-269/270
and 50-287

JUL 14 1977

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Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

By letter dated March 10, 1977, you requested an exemption from the provisions of Appendix H to 10 CFR 50 which would permit future operation of the Oconee Nuclear Station Units Nos. 1, 2 and 3 while irradiating the remaining reactor vessel surveillance specimens at Crystal River, Unit 3. This exemption was requested because damage to surveillance capsule holders prevented the surveillance program from being conducted at Oconee without substantial radiation exposure to plant personnel. By the same letter, you requested approval of proposed changes to the Oconee common Technical Specifications consistent with the requested exemption.

Irradiating the remaining Oconee surveillance specimens at Crystal River Unit 3 or in test reactors will cause the Oconee program to be out of conformance with the provision of Appendix H which requires the irradiation program to be performed within the Oconee vessels. However, as noted in the enclosed Safety Evaluation, the nominal dimensions of Oconee reactor vessels and internals are identical to those at Crystal River Unit 3, and the operating conditions at these two reactors are very similar so that with the exception of actual operating history and minor differences in power distribution, for which adjustments can be provided, the technical aspects of the material surveillance program will be achieved to our satisfaction.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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Based on these considerations, we have concluded that an exemption for Oconee Units Nos. 1, 2 and 3 from this requirement for a continuing in-vessel material surveillance program as set forth in Appendix H to 10 CFR Part 50 for a period of five years will not be detrimental to life or property or the common defense and security and is in the public interest. Therefore, the exemption requested in your letter of March 10, 1977, is approved for a period of five years from the date of this letter.

If an extension of this exemption beyond this initial five year term is desired, you should submit an application for extension to the Commission no later than six months prior to expiration of the exemption. This application should provide a justification for extending the term of the exemption based on operating experience.

In addition to granting this exemption, the Commission has issued the enclosed Amendments Nos. 44, 44, and 41 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1,a2 and 3. The amendments provide for performance of the Oconee reactor-vessel material surveillance program at Crystal River Unit No. 1 and for the submission of specified reports. Certain changes were required in the proposed Technical Specifications submitted by your staff relative to this program. These have been discussed with and agreed to by your staff.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 44 to DPR-38
2. Amendment No. 44 to DPR-47
3. Amendment No. 41 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

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DATE ➤	6/ /77	6/ /77	6/ /77	6/ /77	6/ /77	6/ /77	

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Micheal McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Coutland Street, N. E.
Atlanta, Georgia 30308

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

DUKE POWER COMPANY

DOCKET NO. 50-295

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendments by Duke Power Company (the licensee) dated March 10, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I:

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. After weighing the environmental aspects involved, the issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by a change to the Techical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-38 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for Charles M. Trammell

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Sepcifications

Date of Issuance: July 14, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-295

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendments by Duke Power Company (the licensee) dated March 10, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I:

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. After weighing the environmental aspects involved, the issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by a change to the Techical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-47 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell
for
A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Sepcifications

Date of Issuance: July 14, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-295

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendments by Duke Power Company (the licensee) dated March 10, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I:

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. After weighing the environmental aspects involved, the issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by a change to the Techical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-55 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell for

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Sepcifications

Date of Issuance: July 14, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 44 TO DPR-38

AMENDMENT NO. 44 TO DPR-47

AMENDMENT NO. 41 TO DPR-55

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

Revise Appendix A as follows:

1. Remove pages 3.1a, 3.1-4, 4.2-2, 4.2-3 and replace with identically numbered pages

2. Add page 4.2-4

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 4.8 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1421[7].

Figures 3.1.2-1A, 3.1.2-2A, and 3.1.2-3 present the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic test respectively. The limit curves are applicable up to the fifth effective full power year of operation. These curves are adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-1A for reactor criticality and on Figure 3.1.2-3 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core areas, or in test reactors.

The limitation on steam generator pressure and temperature provide protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure if achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure. The limitations of 110°F and 237 psig are based on the highest estimated RT_{NDT} of +40°F and the preoperational system hydrostatic test pressure of 1312 psig. The average metal temperature is assumed to be equal to or greater than the coolant temperature. The limitations include margins of 25 psi and 10°F for possible instrument error.

The spray temperature difference is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

Bases Units 2 and 3

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. (1) These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100°F per hour satisfies stress limits for cyclic operation. (2) The 237 psig pressure limit for the secondary side of the steam generator at a temperature less than 110°F satisfies stress levels for temperatures below the DTT. (3) The reactor vessel plate material and welds have been tested to verify conformity to specified requirements and a maximum NDTT value of 20°F has been determined based on Charpy V-Notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40°F.

Figures 3.1.2-1B and 3.1.2-2B contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT(4) and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4.10.(4) The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this or a similar reactor vessel or in test reactors.(5) The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 MeV) exposure of the reactor vessel is 3.0×10^{10} n/cm^2 -- s at 2,568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation. (6) The calculated maximum values are 2.2×10^{10} n/cm^2 -- s and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load. (4) Figure 3.1.2-1B is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1B is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days, which is equivalent to 655 days at 2,568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of this or a similar vessel in the core area or in test reactors. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

Amendments Nos. ~~38~~, ~~38~~ & ~~35~~
44, 44 & 41

4.2.3 The structural integrity of the Reactor Coolant System boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated, including evaluation of comparable areas of the Reactor Coolant System.

4.2.4 The results of the Inservice Inspections performed pursuant to Specifications 4.2.1, 4.2.2, and 4.2.3 shall be reported to the Commission within 90 days of completion.

4.2.5 To assure the structual integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.6 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.7 The inservice inspection program shall be reviewed at the end of five years to consider incorporation of new inspection techniques and equipment which have been proved practical and the conclusions of this review and evaluation shall be discussed with the NRC.

4.2.8 At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspection is greater than 6 2/3 years.

4.2.9 The reactor vessel material irradiation surveillance specimens removed from Units 1, 2 and 3 reactor vessels in 1976 shall be installed, irradiated in and withdrawn from the Crystal River Unit 3 reactor vessel in accordance with the schedule shown in Table 4.2-1. Following withdrawal of each capsule listed in Table 4.2-1, Duke Power Company shall be responsible for testing the specimens in those capsules and submitting a report of test results in accordance with 10 CFR 50, Appendix H.

4.2.10 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of the following: After March 13, 1978, any time that Crystal River Unit No. 3 fails to maintain a cumulative reactor utilization factor of greater than 45%.

The report shall provide justification for continued operation of Oconee Nuclear Station Units 1, 2 and 3 with the reactor vessel surveillance program conducted at Crystal River Unit No. 3 or the application for license amendment shall propose an alternative program for conduct of the reactor vessel surveillance program.

4.2.11 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspceted along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

4.2.12 To assure that reactor internals vent valves are not opening during operation, all vent valves will be inspected during each refueling outage to confirm that no vent valve is stuck open and that each valve operates freely.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

For the purpose of Technical Specification 4.2.10, Cumulative reactor utilization factor is defined as: [(Cumulative thermal megawatt hours since attainment of commerical operation at 100% power) x 100] ÷ [(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)]. The definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commerical operation".

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.

TABLE 4.2 - 1

OCONEE NUCLEAR STATION CAPSULE ASSEMBLY

WITHDRAWAL SCHEDULE AT CRYSTAL RIVER UNIT NO. 3

Capsule Designation	Insertion	Withdrawal
OCI-A	End of 1st Cycle	End of 7th Cycle
OCI-B	End of 7th Cycle	End of 16th Cycle
OCI-C	End of 2nd Cycle	End of 11th Cycle
OCI-D	End of 9th Cycle	End of 18th Cycle
OCII-A	End of 1st Cycle	End of 2nd Cycle
OCII-B	End of 4th Cycle	End of 9th Cycle
OCII-D	End of 9th Cycle	End of 18th Cycle
OCII-E	End of 1st Cycle	End of 9th Cycle
OCII-F	End of 9th Cycle	End of 18th Cycle
OCIII-B	End of 1st Cycle	End of 2nd Cycle
OCIII-C	End of 2nd Cycle	End of 7th Cycle
OCIII-D	End of 1st Cycle	End of 9th Cycle
OCIII-E	End of 5th Cycle	End of 18th Cycle
OCIII-F	End of 11th Cycle	End of 20th Cycle

Note:

OCI-__ Capsules are from Unit No. 1

OCII-__ Capsules are from Unit No. 2

OCIII-___ Capsules are from Unit No. 3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 44 TO FACILITY LICENSE NO. DPR-38

AMENDMENT NO. 44 TO FACILITY LICENSE NO. DPR-47

AMENDMENT NO. 41 TO FACILITY LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNIT NOS. 1, 2 AND 3

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

Introduction

By letter dated March 10, 1977, Duke Power Company (licensee) requested that (1) an exemption to Appendix H of 10 CFR 50 be granted for the Oconee Units 1, 2 and 3 which would allow indefinite operation of the Oconee Units with the remainder of the reactor vessel surveillance capsules to be irradiated at Crystal River, Unit 3 rather than at Oconee Station, and (2) the Oconee Nuclear Station Technical Specifications be revised to allow the Oconee 1, 2 and 3 reactor vessel surveillance capsules to be irradiated at Crystal River, Unit No. 3. Several modifications to the licensees proposal were necessary. These changes were discussed with and agreed to by the licensee. In addition, information submitted by letter dated January 4, 1977 was considered in this evaluation.

The basis for this request is that the surveillance specimen holder tubes in Oconee were damaged and could not be repaired without a complex and expensive repair program and radiation exposure to personnel. In lieu of putting the surveillance capsules back into the Oconee reactor pressure vessels, they will be placed in a host reactor, Crystal River Unit No. 3, for irradiation. In addition, data from specimens from other irradiation programs will provide input to the Oconee irradiation program. This overall program is called an integrated surveillance program in which all presently operating facilities with B&W 177 fuel assembly reactors will participate.

Discussion & Evaluation

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.

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 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 or will start initial operation within the next few months. All of these plants have the same basic B&W 177 fuel assembly reactor 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 NRC 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 radiological difficulties because installation was 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 installation 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 reinstall the SSHTs in Oconee 1, 2 and 3, they would be likely to cause significant radiation to personnel. Based on its experience in removing the SSHTs at Three Mile Island-1 and Rancho Seco-1, B&W estimated that installing SSHTs in irradiated reactors could result in personnel exposures totaling 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.

The capsules removed from the Oconee vessels which had damaged SSHTs will be placed in a host reactor, Crystal River Unit 3, as part of the integrated surveillance program discussed herein. These capsules contain samples of plate or forging material and heat-affected zone material from the vessel beltline as well as weld metal. The weld metal is expected to be controlling because it is more radiation sensitive. However, capasules containing other than weld metal will be irradiated also, since the purpose of the surveillance program is to obtain data on materials which would prove to be important later on.

This program 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 program at Oconee and other plants with 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 this plan.

1. 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 significant radiation exposure of the workmen, and

2. 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 the beltline welds of a given reactor some of which will come from other surveillance programs, 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 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 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 relevant to Oconee 1, 2 and 3 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 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 x 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 azimuthal locations of these longitudinal welds is lower (0.7 to 1.0 x 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. This 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

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

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.

Oconne 3

There are three weld materials of primary interest for the Oconee 3 vessel, 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 x 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.O. 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.

Table 4.2-1 of the proposed Technical Specifications specifies the Oconee specimens capsules that are to be irradiated in Crystal River Unit No. 3. These capsules include the weld material shown in Table 2 herein and other materials such as plate or forging material samples and weld heat affected zone material samples from the Oconee vessels which are not now considered to be controlling material but could become so sometime in the future.

For those welds where no surveillance specimens exist, we will be guided by predictions based on the known chemical composition of those welds. To be conservative, the predictions will be based on the Upper Limit lines of Regulatory Guide 1.99, Revision 1.

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 irrdiated 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.

We have evaluated the effectiveness of this overall program plan, and have 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 our opinion that even without additional irradiation surveillance programs in Oconee vessels, 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, we intend to use the Upper Limit lines as the basis for any prediction required at this time.

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

TABLE 1

RADIATION DATA FOR

OCONEE - 1 REACTOR

Weld	Capsule*	Host Reactor	Removal Date	Specimen* Types
WF-25 (Upper Circumferential, 39%, on OD)	TMI-1E TMI-1A TMI-1C R-1 R-2 HSST-2 HSST-3 NRL NRL	TMI-1 TMI-2 TMI-2 TMI-2 TMI-2 Test Reactor Test Reactor Test Reactor Test Reactor	1976 1982 1990 1982 1989 1977-78 1978 1977-78 1977-78	Cv, Tensiles Cv, Tensiles Cv, Tensiles Cv, CT Cv, CT Cv, CT to 4.0T Cv,CT Cv,CT
SA-1229 (Upper Circumferential, 61% on ID)	none			
SA-1585 (Center Circumferential)	R-1 R-2 HSST-3	CR-3 CR-3 Test Reactor	1982 1989 1978	Cv, CT Cv, CT 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)				

*TMI-IE - means capsule E in the Three Mile Island Unit No. 1 Reactor

TABLE 2

RADIATION DATA FOR

OCONEE -2 REACTOR

Weld	Capsule *	Host 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
	HSST-3	Test Reactor	1978	4.0T
	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	1976	Cv, tensile
	OCI-A	Crystal R.3	1985	Cv, tensile
	OCI-C	Crystal R.3	1989	Cv, tensile
	R2	Davis-Besse	1989	Cv, CT

*OCI-E - means capsule E from the Oconee Unit No. 1 reactor

**Cv - means Charpy V-notch specimen
CT - means Compact Tension specimen

TABLE 3

RADIATION DATA FOR

OCONEE - 3 REACTOR

Weld	Capsule *	Host Reactor	Removal Date	Specimen ** Types
WF-67 (Center Circumferen-tial 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 Circumferen-tial 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 Circumferen-tial Weld)	none			
WF-169 (Lower Circumferen-tial Weld)	none			

*R-1, R-2 - denotes "research" capsule
** See Table 2

We have also 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.

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 ± 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.

   We have 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 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 177 fuel assembly reactor to another are claimed to be relatively minor, because they have almost identical geometry.

We considered the possible differences in neutron spectra that could occur between the B&W power reactors to be 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 can 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 proposed 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 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.

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 alternative 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, our 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, we intend to use the Upper Limit lines as the basis for any prediction required at this time.

Environmental Consideration

We have determined that these amendments do 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 these amendments involve 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, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 14, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 44 , 44, and 41 to Facility Operating License Nos. DPR-38, DPR-47, and DPR-55, respectively, issued to Duke Power Company which revised the Technical Specifications for operation of the Oconee Nuclear Station, Unit Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

The amendments authorized changes in the Technical Specifications to permit irradiation of the remaining Oconee Nuclear Station, Units Nos. 1, 2 and 3 reactor vessel surveillance specimens at Crystal River Unit 3. An exemption to that provision of Appendix H to Title 10 of the Code of Federal Regulations Part 50, which would have otherwise required irradiation of the Oconee capsules in the Oconee vessels, has also been granted. Such action was in response to a generic failure of first-generation design Surveillance Specimen Holder Tubes (SSHTs) at Oconee and other operating Babcock & Wilcox 177 fuel assembly reactors.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statment or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated March 10, 1977, (2) Amendments Nos. 44, 44 and 41 to License Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington D. C. and at the Oconee County Library, 201 South Spring Street, Walhalla, South Carolina 29691. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of July 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell

Charles M. Trammell, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors