REGULATORY DOCKET FILE COPY

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

November 7, 1980

Dockets Nos. 50-269, 50-270
and 50-287

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

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 88 , 88 , and 85 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your requests dated October 1, 1976, and July 8, 1977, as combined in your request dated May 30, 1979, and supplemented May 26, 1977, September 21, 1977, June 11, 1979, and March 24, 1980.

These amendments revise the Technical Specifications relating to the Inservice Inspection Program, in accord with 10 CFR 50.55a(g).

We have found the program in compliance to the extent possible with the requirements set forth in Section XI of the 1974 Edition and Addenda through the Summer 1975 of the ASME Boiler and Pressure Vessel Code. We have evaluated requests and by past actions granted interim relief from Code requirements. This action hereby grants permanent relief of certain of your requests from specific requirements which were determined to be impractical for the facility because of limited access due to design and radiation, geometry and materials of construction of some components as discussed in the enclosed Safety Evaluation. We have determined that the granting of this relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. A single relief request affecting all three units regarding Authorized Nuclear Inspectors not being licensee employees is denied for reasons discussed in the Safety Evaluation.

8012020036

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 88 to DPR-38
2. Amendment No. 88 to DPR-47
3. Amendment No. 85 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

Duke Power Company

cc w/enclosure(s):

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cc w/enclosure(s) & incoming dtd.:
10/1/76, 7/8/77, 5/30/79, 5/26/77,
9/21/77, 6/11/79 & 3/24/80
Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. DPR-38

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

   A. The applications for amendment by the Duke Power Company (the licensee) dated October 1, 1976, and May 30, 1979, as supplemented May 26, 1977, and June 11, 1979, and application dated March 24, 1980, comply 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 applications, 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. 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 changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR- 38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 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

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 7, 1980

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.8
License No. DPR-47

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

   A. The applications for amendment by the Duke Power Company (the licensee) dated July 8, 1977, and May 30, 1979, as supplemented September 21, 1977, and June 11, 1979, and application dated March 24, 1980, comply 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 applications, 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. 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 changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 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

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 7, 1980

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. DPR-55

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

   A. The applications for amendment by the Duke Power Company (the licensee) dated July 8, 1977, and May 30, 1979, as supplemented September 21, 1977, and June 11, 1979, and application dated March 24, 1980, comply 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 applications, 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. 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 changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 85 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

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 7, 1980

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 88 TO DPR-38

AMENDMENT NO. 88 TO DPR-47

AMENDMENT NO. 85 TO DPR-55

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

Revise Appendix A as follows:

Remove Pages	Insert Pages
iii	iii
vi	vi
4.2-1	4.2-1
4.2-2	4.2-2
4.2-3	4.2-3
4.2-4	--
4.3-1	4.3-1

LIST OF TABLES

4.2 STRUCTURAL INTEGRITY OF ASME CODE CLASS 1, 2 AND 3 COMPONENTS

Applicability

Applies to the surveillance of the ASME Code Class 1, 2 and 3 components.

Objective

To assure the continued structural integrity of the ASME Code Class 1, 2 and 3 components.

Specification

4.2.1 Inservice examination of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g)(4), to the extent practicable within the limitations of design, geometry and materials of construction of the components, except where specific written relief has been granted by the Commission.

4.2.2 To assure the structural 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.

4.2.3 At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an inplace, 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.

4.2.4 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.

4.2.5 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 alternate program for conduct of the reactor vessel surveillance program.

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10 CFR 50.55(a) to the extent practicable within limitations of design, geometry and materials of construction. 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 4.2.5. Cumulative reactor utilization factor is defined as: {(Cumulative thermal megawatt hours since attainment of commercial 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 "commercial operation".

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 5th Cycle	End of 10th 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

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2200 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes. The specific code and edition thereof shall be consistent with 10CFR 50.55.

REFERENCES

FSAR, Section 4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

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

Introduction and Background

By letters dated October 1, 1976 and July 8, 1977, as combined in a letter dated May 30, 1979, and as supplemented May 26, 1977, September 21, 1977, and June 11, 1979, the Duke Power Company (DPC or the licensee) proposed changes in Technical Specification (TS) 4.2-Reactor Coolant System Surveillance. TS 4.2 is the Oconee Nuclear Station (ONS) Inservice Inspection (ISI) Program. By letter dated March 24, 1980, DPC proposed a change to TS 4.3.2 regarding leak test pressure of the reactor coolant system (RCS). The proposed changes are designed to assure the Station is in compliance with 10 CFR 50.55a(g); in order to meet this Regulation, the ISI Program should be in compliance to the extent practicable with the 1974 Edition and Addenda through Summer 1975 of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The ONS was designed and constructed prior to the implementation of the Code, and as a consequence, due to the system design, geometry, or materials of construction of the components, could not meet, in every case, the later Code requirements. However, the Regulation and the Code provide for relief from these requirements if impractical. Relief may be granted if acceptable alternate inspection requirements are proposed. Stated below are specific relief requests on a reactor unit by unit basis and our evaluation of each relief request. The licensee submittals of May 26, 1977, and September 21, 1977, constitute the relief requests specific to Units 1, 2 or 3; the June 11, 1979, submittal is a relief request common to all three units of the Station and is evaluated separately.

By letters dated January 18, 1978, for ONS Unit 1 and March 27, 1978, for Units 2 and 3, the NRC granted interim relief to all of the licensee's requests except the request in Duke's letter of June 11, 1979, which is evaluated herein. These interim reliefs permitted the three units to perform an ISI program that met the requirements of 10 CFR 50.55a(g) pending the completion of our detailed review as provided by this Safety Evaluation.

I. CLASS 1 COMPONENTS

A. Reactor Vessel

1. Relief Request

Relief from reactor pressure vessel nozzle inspection (Category B-D) as required by paragraph IWB-2411 is requested for ONS Units 1, 2 and 3.

Code Requirement

At least 4 of 8 nozzles must be inspected by completion of 80 months of operation.

Licensee Basis for Requesting Relief

The net effect of the above Code requirements is that four nozzles of a total of eight must be examined by the end of the 80 months of commercial operation. Due to core support structure design only the two reactor coolant outlet nozzles are accessible without removing the core barrel, which in turn requires complete defueling. This requirement is considered to be impractical. In lieu of the above, the following examination sequence is proposed:

	Components to be Examined	Examinations Schedule (Elapsed Time Since Commercial Service Date)
1	Reactor Coolant Outlet Nozzle	Approximately 40 months[1]
1	Reactor Coolant Outlet Nozzle	Approximately 80 months[1]
4	Reactor Coolant Inlet Nozzles	Approximately 120 months
2	Core Flooding Nozzles	Approximately 120 months

[1] Different nozzle will be examined each inspection period.

Evaluation

Examination of the nozzles in the sequence required by the Code places an undue burden on the licensee because examination of the reactor coolant inlet and core flooding nozzles can only be accomplished by removing the fuel and core barrel from the reactor vessel. Examination of the outlet nozzles as proposed by the licensee will constitute a representative sample of the condition of the remaining nozzles because of similar service conditions. If any unacceptable flaw is found during an examination of the outlet nozzles, the examination will have to be extended to include additional nozzles as required by the Code.

We find the proposed sequence for examination of the nozzles acceptable, will provide assurance of their structural integrity, and will not significantly decrease the safety of the facility. We conclude that relief from the Code requirement may be granted as compensated for by the licensee's proposed schedule.

2. Relief Request

Relief from the requirements of Table IWB-2500 Category B-I-1 and paragraph IWB-2411 for the reactor vessel clad patch examination is requested for ONS Units 1, 2 and 3.

Code Requirement

Except as specified by IWB-2500 for examinations that may be deferred to the end of the inspection interval, at least 25% of the required examination shall be completed by the expiration of one-third of the inspection interval and at least 50% shall be completed by the expiration of two-thirds of the inspection interval.

Licensee Basis for Requesting Relief

Performance of these examinations requires complete defueling of the core and removal of core barrel.

Evaluation

The licensee has proposed to inspect 100% of the required cladding patches at the end of the inspection interval. The inspection sequence does not comply exactly to Code requirements. The licensee's examination of the reactor vessel outlet nozzles will cover sufficient area of the cladding to indicate the cladding general condition. We conclude that the safety gained by the Code inspection sequence requirements exactly is not commensurate with the burden placed upon the licensee. We conclude that the proposed inspection sequence is acceptable and relief from Code requirements for the reactor vessel cladding inspection sequence may be granted.

B. Piping Pressure Boundary

1. Relief Request

Relief from the requirements of Table IWB-2600, Item 4.9 Category B-k-1 is requested for:

Oconee Unit No. 1 Low Pressure Injection and Core Flooding Systems' Welds Nos. 46LA and 60LB.

Oconee Unit No. 2 Core Flood Tank and Decay Heat Removal Systems' Welds Nos. 53A and 102A and High Pressure Injection System Welds 932 and 89C.

Code Requirement

The welds of external support attachments to the pressure retaining boundary including the base metal beneath the weld zone and along the support member for a distance of two support thicknesses shall be volumetrically examined.

Licensee Basis for Requesting Relief

The weld geometry of the attachment welds prevents meaningful volumetric examination.

Evaluation

Because of the weld design, a meaningful ultrasonic examination cannot be accomplished.

Radiographic examination of these welds would be difficult to perform and interpret, and would therefore result in little added assurance to safety. The licensee has committed to subject these welds to surface examination. Based on the loading conditions of these types of welds, flaws would most likely generate at the weld surface and thus be detectable by surface examination.

We requested that the licensee also perform an ultrasonic examination of the base metal in order to assure that flaws in the base metal do not exist. The licensee agreed to perform this added examination. The combination of surface examination and recommended volumetric base metal exam would provide assurance that the integrity of the pipe supports would be maintained.

II. CLASS 2 COMPONENTS

1. Relief Request

Relief is requested from system pressure tests for ONS Unit 1 Main Steam System Weld No. 50 which is to be replaced by a 1 inch socket weld on the inlet side of a manually operated steam drive valve.

Code Requirement

IWA-4210, Pressure Test, requires that after repair by welding on the pressure retaining boundary of components, a pressure test shall be performed in accordance with the requirements of IWA-5000 and IWC-5000 for Class 2 weld.

Licensee Basis for Requesting Relief

The weld in question is not directly on the main steam header. Hydrostatic testing of the weld would require pressurizing the steam generator secondary side, main steam lines and sections of the feedwater header. It would also require heatup of the steam generator and involve operation of many related systems. It is estimated that 7-8 days of down time would be required to perform this hydro. Additionally, the potential for damage to the main steam system is high since it is not designed to be filled with water. With one weld involved, it is felt that a system leak test at operating conditions is as reliable as hydrostatic test to assure leak tightness. In addition, an examination using the liquid dye penetrant technique along with ultrasonic testing will be performed.

Evaluation

From the piping diagrams, it is clear that the licensee would have to pressurize the connected systems in order to hydrotest this repaired weld. The licensee has proposed an inservice leak test at operating pressure and temperature in addition to the surface and volumetric examinations to be performed. Considering the one inch nominal pipe size system with which this weld is associated, we believe that the proposed alternative testing will ensure the integrity of the weld and this relief request may be granted. However, it is noted that the hydrotest required by the Code during each inspection interval is in no way waived by the granting of this relief request.

2. Relief Request

Relief from hydrotesting the decay heat removal cooler outlet control valves LP-12 and LP-14 to 125% of design pressure after valve replacement is requested for ONS Units 1, 2 and 3. The licensee proposes instead to test the valve to 100% of design pressure on upstream and downstream sides and conduct radiography of 100% of the valve joint welds.

Code Requirement

The system hyrostatic test pressure shall be at least 1.25 times the system design pressure.

Licensee Basis for Requesting Relief

The decay heat removal coolers are located upstream of these valves. The piping between the coolers and the valves is designed for 350 psig at 300°F. The valves are not leak tight, having design leakage of 0.5% and being normally used to control flow, not for isolation purposes. The piping downstream of the valves is designed for 505 psig at 250°F. Upon replacement of these valves, the welded joints were required to be hydrostatically tested to 125% of design pressure. Even the piping downstream of the valves would be tested to normal operating pressure and with leakage through the valves, the piping with lesser design pressure as well as the coolers could become overpressurized and possibly damaged. Venting of the upstream piping to relieve the pressure buildup could release significant amounts of high activity waste and prevent satisfactory completion of the hydrostatic test.

Evaluation

The circumstances which surround this particular test clearly prevent weld isolation such that the Code required pressure can be used in performing this test. The alternative testing proposed will assure, with the supplemental radiographic techniques, that any flaws developed during repair or service will be found. Additionally the weld will be hydrostatically tested to 100% of design pressure and thus will provide reasonable assurance that large flaws will be detected. We find that the basis for this request is reasonable and that relief from hydrotesting the repair welds to 125% of design pressure may be granted.

III. CLASS 3 COMPONENTS

1. Relief Request

Relief from pressure testing to 1.10 times the design pressure the reactor coolant pump seal supply line is requested for ONS Unit 1. This line contains welds 1K and 1L which are new welds installed during a Station modification performed during the 1977 shutdown.

Code Requirement

Visual examination shall be conducted for evidence of component leakage, structural distress, or corrosion when the system is undergoing a system pressure test. The system test pressure shall be at least 1.10 times the system design pressure.

Licensee Basis for Requesting Relief

The design pressure for this system is 3050 psig. In order to meet the requirements of the Code, the test pressure must be 3355 psig. The test pressure exceeds that pressure allowed with fuel in the vessel. The new welds, 1K and 1L, were radiographed and found acceptable during the 1977 refueling shutdown.

Evaluation

The design of the reactor coolant pump seal supply line prevents isolation of the seal supply line for pressure testing at a pressure specified by the Code. The repair welds have been radiographed and found acceptable. In addition, we requested that the seal supply line be visually examined during a hydrostatic test at 100% of system pressure. DPC agreed to include these two welds in the hydrotest program. We believe that the combination of radiography and recommended pressure testing will provide adequate assurance of system integrity and this request may be granted.

IV. GENERAL

1. Relief Request

Relief from the rules of ASME Section XI IWA-2120 regarding the Authorized Nuclear Inspectors witnessing or auditing test results is requested for ONS Units 1, 2 and 3.

Code Requirement

IWA-2130(b) states that any inspector who performs inspections required by this Division shall have first been qualified by written examination pursuant to the legislation of rules of a state of the United States, the legislation of a Canadian Province, or the rules of another authority having jurisdiction over a nuclear power plant at the installation location and that has adopted this Division. The Inspector shall not be an employee of the owner or his agent.

Licensee Basis for Requesting Relief

The duties of the Authorized Inspector as stated in ASME Code Section XI, IWA-2120, are performed to the full extent by personnel within the Quality Assurance Department. This department of DPC is organizationally separate from those persons responsible for performing engineering, construction, or operating functions. The personnel within the Quality Assurance Department have the required independence and authority to effectively carry out the quality assurance program without undue influence from those directly responsible for costs and schedules.

Evaluation

IWA-2130(b) states specifically that the Inspector shall not be an employee of the owner or his agent. It is concluded that the licensee should meet the requirements of the Code concerning the Authorized Nuclear Inspector. The purpose of the requirement is to have an independent third party as the Inspector. The issue was discussed with DPC and they agreed that an independent third party Inspector will be used.

2. Relief Request

Relief from the holding time requirement for system hydrostatic and leak tests (IWA-5210) is requested for ONS Units 1, 2 and 3.

Code Requirement

The pressure-retaining components shall be visually examined while the system is under the hydrostatic test pressure and temperature. The test pressure and temperature shall be maintained for at least four hours prior to the performance of the examinations.

Licensee Basis for Requesting Relief

The approved Code - 1974 Edition including through Summer 1975 Addenda - contains in Section XI, Article IWA-5000, System Pressure Test, the requirement to perform a hydrostatic test "for at least four hours prior to the performance of the examinations" (IWA-5210(a)). The performance of this test is impractical particularly in situations where boundary valves used to isolate piping sections for this test are not designed for zero leakage and leak by the seats at a rate greater than the capacity of the hydrostatic pump. In addition, it is not necessary to wait four hours to observe leakage of a welded joint. Any weldment leakage would be readily observable within a matter of minutes.

In a later edition of the Code - 1977 Edition including through Summer 1978 Addenda - the hydrostatic test requirements are stated for conditions of insulated and noninsulated piping. The four-hour hydrostatic test is required only for insulated systems while only 10 minutes is required for a noninsulated piping system. However, this edition of the Code is not currently approved by the NRC.

It is the position of DPC that this later edition of the Code clarifies the previous edition of the Code with respect to the four-hour hydrostatic test and is applicable in cases where hydrostatic testing is required.

Evaluation

The four-hour holding time required by the 1974 Edition of Section XI during hydrostatic tests is intended for application to systems where the base material and weld deposits are covered by insulation. The purpose of the holding time is to allow pressure boundary leakage to become evident at the insulation surface. Where the base material and weld are visible, the intent of the holding time is meaningless and deletion of this requirement will not decrease the effectiveness of the examination. We conclude that this request may be granted with the following conditions, which have been discussed with and agreed to by DPC:

a) When performing a system pressure test the entire system must be visible directly. This includes the welds and all base materials.

b) When the areas are exposed, the pressure and temperature required by the Code for the hydrostatic and leak test shall be maintained for a minimum time of ten (10) minutes and for such additional time as may be necessary to conduct the examinations.

c) Following a repair, the repaired area must be accessible for a direct visual examination.

Conclusions on Relief Requests and ISI Program

We find that the proposed Technical Specification changes are acceptable in that they meet the requirements of 10 CFR 50.55. The licensee has submitted information to support his determinations that certain ASME Section XI Code (1974 Edition through Summer 1975) requirements are impractical to implement at ONS. We have evaluated the licensee's bases for his determinations and find that relief from the specific Code requirements requested may be granted except for using a DPC employee in the position of an Authorized Nuclear Inspector. We conclude that the revised ISI Program meets the requirements of 10 CFR 50.55a(g).

V. LEAK TEST PRESSURE FOLLOWING OPENING OF REACTOR COOLANT SYSTEM (RCS)

By letter dated March 24, 1980, DPC requested that the leak testing pressure following any opening of the RCS be reduced from 2285 psig to 2200 psig. Section XI of the ASME Code in Paragraph IWB-5221 requires that the system leak test pressure be no less than the system nominal operating pressure at 100% rated reactor power. The Final Safety Analysis Report states that the nominal operating pressure is 2150 psig; we have verified that all three ONS units currently (1980) operate at 2150 psig. As the requested leak test pressure of 2200 psig is greater than the Code minimum of 2150 psig, we conclude the change is acceptable.

Environmental Consideration

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

Conclusion

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

Dated: November 7, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES AND GRANTING OF RELIEF FROM ASME SECTION XI INSERVICE INSPECTION REQUIREMENTS

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 88, 88 and 85 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company (the licensee), which revised Technical Specifications for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

The amendments replace the current inservice inspection Technical Specifications with an inservice inspection program that meets the requirements of 10 CFR 50.55a.

By letter dated November 7, 1980, as supported by the related Safety Evaluation, the Commission has also granted to the licensee relief from certain requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components". The relief relates to the inservice inspection program for the Station. The ASME Code requirements are incorporated by reference into the Commission's rules and regulations in 10 CFR Part 50. The relief is effective as of its date of issuance.

The applications for the amendments and requests for relief comply 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, and letter granting relief. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendments dated October 1, 1976, and July 8, 1977, as combined in the application dated May 30, 1979, and as supplemented May 26, 1977, September 21, 1977, and June 11, 1979, and the application dated March 24, 1980, (2) Amendments Nos. 88 ,88 and 85 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, (3) the Commission's related Safety Evaluation, and (4) the Commission's letter to the licensee dated November 7 ,1980. 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. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 7th day of November 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing