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Am-27 to

DPR-66

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

April 7, 1980

Docket No. 50-334

Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Dear Mr. Dunn:

By letters dated February 15 and 21, 1980 you requested additional relief from ASME Code requirements and committed to additions, respectively, to your previously approved Beaver Valley Nuclear Power Station, Unit No. 1 Inservice Inspection (ISI) Program. You also requested deletion of all references to Table 4.4-4 from the plant's Appendix A Technical Specifications on the basis that NRC approval of these requests and our earlier December 4, 1979 approval of your ISI program would make this table obsolete.

For the reasons set forth in the enclosed Safety Evaluation, we hereby grant the relief from ASME Code requirements requested in your February 15, 1980 letter and approve the additions to your ISI program which your February 21, 1980 letter commits.

The relief from ASME Code requirements granted by the Safety Evaluation shall remain in effect until specifically revoked by the NRC or until the end of the 120-month period beginning September 30, 1976.

Table 1 of the enclosed Safety Evaluation delineates those items for which relief has been granted and alternative examinations defined. We are granting this relief and imposing alternative examinations based on our review of the information you submitted to support your determinations that certain ASME Code requirements would be impractical for your facility. 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 will otherwise be in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on your facility.

The Commission has also issued the enclosed Amendment No. 27 to Facility Operating License No. DPR-66 for the Beaver Valley Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated February 15, 1980 as supplemented February 21, 1980.

The amendment deletes all references to Technical Specification Table 4.4-4. This deletion is administrative in nature since the ISI requirements formerly

covered by that table are now provided in the NRC approved Beaver Valley, Unit No. 1 ISI program as amended. The notation to Table 4.4.4 is retained in Technical Specification 3/4.4.13 however, until that program is complete.

A copy of the related Federal Register Notice is also enclosed.

Sincerely,

A. Schwencer

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

Enclosures:
1. Amendment No. 27 to DPR-66
2. Safety Evaluation
3. Federal Register Notice

cc: w/enclosures
See next page

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees) dated February 15, 1980 as supplemented February 21, 1980, 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. 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 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 27, 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

A. Schwencer

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 7, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
V	V
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3/4 4-29	3/4 4-29
3/4 4-30a	--
3/4 4-30b	--
3/4 4-30c	--
3/4 4-30d	--
3/4 4-30e	--
3/4 4-30f	--
3/4 4-30g	--
3/4 4-30h	--
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	--

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PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during steady state operation.

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5 and by performing the following augmented inservice inspection program:

SURVEILLANCE REQUIREMENTS (Continued)

a. At the first refueling outage, dye penetrant inspections shall be performed on all six reactor vessel nozzles and volumetric inspections shall be performed on at least two nozzles in the same manner as that conducted in the baseline inspection.

b. At the second and third refueling outages dye penetrant inspections shall be performed on all six nozzles and volumetric inspections shall be performed on at least two nozzles in the same manner as the baseline inspections so that all six nozzles shall be volumetrically inspected during the first three refueling outages.

c. Defects found during these inspections shall be evaluated to determine their significance and reported to the Commission pursuant to Technical Specification 6.9.1.8.c.

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TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	CU %	P %	NDTT °F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	RT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS
CL. HD Dome	A533,8,Cl.1			-40	40	77*	17	115	75**
CL. HD. SEG.	A533,8,Cl.1			-20	18	59*	-1	118	77**
HD. FLG.	A508,Cl.2			10	-25	-5*	10	100	65**
VESSEL FLG.	A508,Cl.2			60	21	56	60	151	98**
INLET NOZZLE	A508,Cl.2			60$	105	200*	140	82	53**
INLET NOZZLE	A508,Cl.2			60$	58	125*	65	94	61**
INLET NOZZLE	A508,Cl.2			60$	72	148*	88	94	61**
OUTLET NOZZLE	A508,Cl.2			60$	12	67*	60	95	62**
OUTLET NOZZLE	A508,Cl.2			60$	98	107*	60	112	73**
OUTLET NOZZLE	A508,Cl.2			60$	55	112*	60	103	67**
UPPER SHELL	A508,Cl.2			40	10	37*	40	150	98**
INTER. SHELL	A533,8,Cl.1	0.14	0.015	-10	57	103	43	123	81
INTER. SHELL	A533,8,Cl.1	0.14	0.015	-10	60	83	23	125	81
LOWER SHELL	A533,8,Cl.1	0.20	0.010	-50	--	87	27	---	75 SURV
LOWER SHELL	A533,8,Cl.1	0.14	0.015	-20	33	80	20	127	75
TRANS. RING	A508,Cl.1			30	30	60*	30	139	90**
BT. HD. SEG.	A533,8,Cl.1			-30	11	46*	-14	122	79**
BT. HD. DOME	A533,8,Cl.1			-50	15	48*	-12	122	79**
WELD	WELD	0.26	0.0.8	-60	--	-16	-60	---	103
HAZ	HAZ			-40	--	20	-40	---	128

* Estimated (77 FT-Lb/54 mil Temp. for Longi.Data)
**Estimated (65% of Longitudinal Upper Shelf)
$ Estimated (60°F or 100 FT-lb Temp. or what is less)
SURV - Material for surveillance program according to ASTM-185-70.

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

Introduction

By letter dated February 15, 1980, Duquesne Light Company (the licensee) requested relief from certain requirements of the 1974 Edition through Summer 1975 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code. This is in addition to relief that was granted by NRC letter dated December 4, 1979.

The licensee also applied in its February 25, 1980 letter for a change to the Appendix A Technical Specification that would eliminate Table 4.4-4 in its entirety.

As discussed below, based on a further revision to The Beaver Valley Inservice Inspection Program as documented in the licensee's letter of February 21, 1980, the Staff is now in agreement that all reference to Table 4.4-4 may be removed from the Technical Specifications as an administrative matter.

In fact, all but the last page of Table 4.4-4 (page 3/4 4-30h) should have been removed from the Technical Specifications on December 4, 1979 when the NRC Staff approved the licensee's October 10, 1977 ISI program, since that approved program now addresses the surveillance requirements of that table.

Discussion and Evaluation

Request for Relief

The licensee requests relief from performing visual and surface or volumetric examination of the reactor vessel closure head cladding and visual examination of the reactor pressurizer, and steam generator vessel cladding.

Code Requirement

Visual and surface or volumetric examination of the reactor vessel closure head cladding shall include at least six patches (each 36 sq. in.) evenly distributed in the closure head. Visual examination of the reactor vessel cladding shall include at least six patches (each 36 sq. in.) evenly distributed in accessible sections of the vessel shell. The examinations performed during each inspection interval shall cover 100% of the patch areas.

Visual examination of the pressurizer and steam generator vessel cladding shall include at least one patch (36 sq. in.) near each manway in the primary side of the vessel. The examinations performed during each inspection interval shall cover 100% of the patch areas. The examination of the patch areas in the pressurizer and steam generator may be performed at or near the end of the inspection interval.

Licensee Basis for Requesting Relief

The design of these vessels accounts for the stress loads to be adequately accommodated by the ferritic base material which is examined volumetrically as required by other examination categories. Additional technical guidance is provided by later editions of Section XI where cladding examinations are no longer required.

Evaluation

Examination of the reactor vessel head cladding patches and the pressurizer and steam generator vessel cladding is impractical to perform because of the relatively high radiation levels present in the areas required to be visually examined. Other examinations which will be performed on these components will give more meaningful data about the structural acceptability of the components. These examinations have been found to be suitable alternatives for the visual examinations of the vessels cladding and the visual and surface or volumetric examination of the vessel head cladding.

Examination of the reactor vessel cladding is also impractical to perform because of the necessity to remove the fuel and core barrel to accomplish the visual examination required. Examination Category B-N-1, which the licensee has committed to perform, requires visual examination of the reactor vessel interior in accessible areas above and below the reactor core during normal refueling outages at approximately three-year intervals. This examination in conjunction with the volumetric examinations performed on shell and nozzel welds will provide adequate assurance of the structural integrity of the reactor vessel.

The Staff finds that the examinations which will be performed by the licensee on the reactor vessel head, reactor vessel, pressurizer, and steam generator will provide data necessary to determine the structural integrity of these components and concludes that relief from the required examinations of the components cladding may be granted as requested.

Table 1 lists the specific items for which we find relief should be granted along with the alternative examinations required. This table supplements Table 1 in The Staff Safety Evaluation dated December 4, 1979 which supports the relief granted at that time from certain other ASME Code requirements. The page number and note number are selected to allow the table in this report to be incorporated into the table in our December 4, 1979 Safety Evaluation Report.

Addition to the Beaver Valley Unit No. 1 ISI Program - Augmented Program for Reactor Vessel Nozzle Safe Ends

The licensee in its February 15, 1980 letter noted that the inspections of the reactor vessel nozzle safe end welds to determine if corrosion is developing in the sensitized zones had been performed in accordance with the Technical Specifications. Therefore, the Notation to Table 4.4-4 (corrected from 4.4-1)

on page 3/4 4-30h of the Technical Specifications (covering three nozzle inspections) should be deleted. However, in discussions with the licensee, it was noted that when the Technical Specifications were first drafted (see Safety Evaluation Report (SER-OL) for the Facility Operating License DPR-66 dated October 1974 as supplemented May 1975 and October 1975), the five year duration cited in the Table 4.4-4 Notation was considered sufficient time for the Beaver Valley Unit No. 1 to have reached its third refueling outage.

Due to a number of technical difficulties, the Unit has only completed one operating fuel cycle and is not scheduled to return to power from its first refueling outage until July 1980. The requirements of the SER-OL are not tied to the five year period but rather to the first refueling and "the following three and one-third year interval coinciding in the scheduled inservice inspections".

By letter dated February 21, 1980, the licensee requested the deletion of the Notation to Technical Specification Table 4.4-4 on the basis that the ISI program will be augmented to include the remaining provisions of the reactor vessel nozzel safe end examinations now specified in that Notation.

The augmented ISI program is to include the following inspections.

1. At the first refueling outage, dye penetrant inspection shall be performed on all six reactor vessel nozzles and volumetric inspection shall be performed on at least two nozzles in the same manner as that conducted in the baseline inspection (completed).

2. At the second and third refueling outages dye penetrant inspections shall be performed on all six nozzles and volumetric inspections shall be performed on at least two nozzles in the same manner as the baseline inspections so that all six nozzles shall be volumetrically inspected during the first three refueling outages.

In the event that defects are indicated during this period of operation, the inspection program shall require an evaluation to determine their significance and shall be reported as required to the Commission pursuant to Technical Specification 6.9.1.8.c.

In the event that defects are not indicated or considered significant during this period of operation, the inspection program shall be performed thereafter in accordance with the requirements of Section XI of the ASME Code.

While we agree that the above augmented program should be included in the Beaver Valley Unit No. 1 ISI program, the Notation to Technical Specification Table 4.4-4 should be maintained in the Technical Specifications pending the examinations following the second and third refueling outages.

TABLE 1
BEAVER VALLEY UNIT 1
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	CODE EXAMINATION REQUIREMENT	ALTERNATIVE EXAMINATION REQUIRED
B1.14	B-I-1	Reactor Vessel Cladding		Internal Cladding	Visual	Note 18
B1.13	B-I-1	Closure Head Cladding		Internal Cladding	Visual and Surface or Volume	Note 18
B2.9	B-I-2	Pressure Vessel Cladding		Internal Cladding	Visual	Note 18
B3.8	B-I-2	Steam Generator Vessel Cladding		Internal Cladding	Visual	Note 18

TABLE 1 NOTES (CONTINUED)

18. General vessel cladding examination will be performed when vessel interiors are accessable. These examinations will be performed once in a ten year interval (consistent with reactor vessel discharge of full core for examination and whenever the pressurizer and steam generator may be made accessable for examination or other repairs).

Environmental Consideration

We have determined that neither the relief requested nor the amendment authorizes a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the relief and the amendment involve an action which is insignficant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this action.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the relief and the amendment do 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, this 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 granting of relief and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 7, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-334

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

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

The U. S. Nuclear Regulatory Commission (the Commission) has granted relief from certain requirements of the ASME Code Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components and has issued Amendment No. 27 to Facility Operating License No. DPR-66 issued to Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees), which revised Technical Specifications for operation of the Beaver Valley Power Station, Unit No. 1 (the facility) located in Beaver County, Pennsylvania. The relief and associated amendment are effective as of

The licensees are relieved from performing visual and surface or volumetric inservice inspections (ISI) examination of the cladding on the reactor vessel head, and visual examination of the cladding in the pressurizer in the steam generators because other examinations have been found to be suitable alternatives.

The amendment is administrative in nature and simply deletes all references to Table 4.4-4. The ISI requirements formerly covered by that table are now provided for in the NRC approved facility ISI program as amended with the exception of the Notation to Table 4.4-4 which is now included in Technical Specification 3/4 4.10.

The request for relief and the application for the amendment 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 letter granting relief and in the license amendment. Prior public notice of this action was not required since neither the relief nor the amendment involves a significant hazards consideration.

The Commission has determined that the granting of this relief and the issuance of this amendment 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 issuance of this action.

For further details with respect to this action, see (1) the licensee's request for relief and application for amendment dated February 15, 1980 as supplemented February 21, 1980, (2) the Commission's letter dated April 7, 1980, (3) Amendment No. 27 to License No. DPR-66 and (4) 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 B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001. 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 Operating Reactors.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
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