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Docket No. 50-331

Iowa Electric Light and Power Company
ATTN: Mr. Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Gentlemen:

RE: DUANE ARNOLD ENERGY CENTER

Minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of water cooled power reactors are specified in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements". This regulation became effective on August 16, 1973 and requires that adequate margins of safety be provided during any condition of normal operation for the pressure boundary over its service lifetime. These conditions include anticipated operational occurrences and system hydrostatic tests. Many nuclear power plants were in advanced stages of design or construction, or had already received operating licenses at the time Appendix G became effective. It was appropriately recognized that these plants would require subsequent evaluation on a case-by-case basis for compliance with Appendix G.

Consequently, you are requested to review the reactor coolant system pressure-temperature limits contained in the Technical Specifications for your facility to determine if they are in full compliance with Appendix G. Your review should include the pressure-temperature limits for heatup and cooldown operations, system hydrostatic tests, and reactor core criticality. If you find that your pressure-temperature limits in your Technical Specifications are not in full compliance with Appendix G, we request that you propose appropriate changes to your Technical Specifications within 60 days.

This request for generic information was approved by GAO under a blanket clearance number B-1980225; this clearance expires July 31, 1977.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Appendix G

cc: See next page

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Appendix G—Fracture Toughness Requirements

I. Introduction and Scope

This appendix specifies minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of water cooled power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi.

B. Welds and weld heat-affected zones in the materials specified in section I.A.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi.

Adequacy of the fracture toughness of other ferritic materials shall be demonstrated to the Commission on an individual case basis.

II. Definitions

A. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, section III, "Rules for the Construction of Nuclear Power Plant Components" (unless another section is specified), 1971 Edition, and addenda through the Winter, 1972 Addenda.[1]

B. "Ferritic material" means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic structure.

C. "System hydrostatic tests" means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, section XI, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems."

D. "Specified minimum yield strength" means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built pursuant to § 50.55a.

E. "Lowest service temperature" means the lowest service temperature as defined by paragraph NB-2332 of the ASME Code.

F. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in paragraph NB-2331 of the ASME Code.

G. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Appendix H) by adding to RT_{NDT} the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material, measured at the 50 ft lb level or measured at the 35 mil lateral expansion level, whichever temperature shift is greater.

H. "Beltline region of reactor vessel" means the shell material (including welds and weld heat-affected zones) that directly surrounds the effective height of the fuel element assemblies and any additional height of shell material for which the predicted adjustment of reference temperature at end of service life of the reactor vessel exceeds 50° F.

I. "Material surveillance program" means the provisions for the placement of reactor vessel beltline material specimens in the reactor vessel, and the program of periodic withdrawal and testing of such specimens to monitor, over the service life of the vessel, changes in the fracture toughness properties of the beltline as a result of exposure to neutron irradiation and the thermal environment.

J. "Integrated surveillance programs" means the combination of individual material surveillance programs as applied to one or more reactor vessels to yield results which serve to monitor the changes in fracture toughness properties for a group of vessels.

III. Fracture Toughness Tests

A. To demonstrate compliance with the minimum fracture toughness requirements of sections IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code, section NB-2300, "Fracture toughness requirements for materials." Both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch test specified by paragraph NB-2321.2 of the ASME Code. In addition, when required by the ASME Code, unirradiated ferritic materials shall be tested by means of the dropweight test specified by paragraph NB-2321.1 of the ASME Code. Provision shall be made for supplemental tests in crucial situations such as that described in Section V.C.

B. Charpy V-notch impact tests and dropweight tests shall be conducted in accordance with the following requirements:

1. Location and orientation of impact test specimens shall comply with the requirements of paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens shall be representative of the actual materials of the finished component as required by the applicable rules of the construction code under which the component is built pursuant to § 50.55a, except that ferritic materials intended for the reactor vessel beltline region shall comply with the additional requirements of section III.C. of this appendix.

3. Calibration of temperature instruments and Charpy V-notch impact test machines used in impact testing shall comply with the requirements of paragraph NB-2360 of the ASME Code.

4. Individuals performing fracture toughness tests shall be qualified by training and experience and shall have demonstrated competency to perform the tests in accord with written procedures of the component manufacturer.

5. Fracture toughness test results shall be recorded and shall include a certification by the licensee or person performing the tests for the licensee that:

a. The tests have been performed in compliance with the requirements of this appendix,

b. The test data are correctly reported and identified with the material intended for a pressure-retaining component,

c. The tests have been conducted using machines and instrumentation with available records of periodic calibration, and

d. Records of the qualifications of the individuals performing the tests are available upon request.

C. In addition to the test requirements of section III.A. of this appendix, tests on ma-

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. N.W., Washington, D.C.

terials of the reactor vessel beltline shall be conducted in accordance with the following minimum requirements:

1. Charpy V-notch (C_v) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_v test curves (including the upper-shelf levels) in terms of both fracture energy and lateral expansion of specimens. Location and orientation of impact test specimens shall comply with the requirements of paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from excess material and welds in the vessel shell course(s) following completion of the production longitudinal weld joint, and subjected to a heat treatment that produces metallurgical effects equivalent to those produced in the vessel material throughout its fabrication process, in accordance with paragraph NB-2211 of the ASME Code. Where seamless shell forgings are used, or where the same welding process is used for longitudinal and circumferential welds in plates, the test specimens may be taken from a separate weldment provided that such a weldment is prepared using excess material from the shell forging(s) or plates, as applicable, the same heat of filler material, and the same production welding conditions as those used in joining the corresponding shell materials.

IV. FRACTURE TOUGHNESS REQUIREMENTS

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials shall meet the following requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

1. The materials shall meet the acceptance standards of paragraph NB-2330 of the ASME Code, and the requirements of sections IV.A.2, 3 and 4 and IV.B. of this appendix.

2. For vessels, exclusive of bolting or other fasteners:

a. Calculated stress intensity factors shall be lower than the reference stress intensity factors by the margins specified in the ASME Code Appendix G, "Protection Against Non-Ductile Failure". The calculation procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture, making appropriate allowance for all uncertainties in the data and analyses.

b. For nozzles, flanges and shell regions near geometric discontinuities, the data and procedures required in addition to those specified in the ASME Code shall provide margins of safety comparable to those required for shells and heads remote from discontinuities.

c. Whenever the core is critical, the metal temperature of the reactor vessel shall be high enough to provide an adequate margin of protection against fracture, taking into account such factors as the potential for overstress and thermal shock during anticipated operational occurrences in the control of reactivity. In no case when the core is critical (other than for the purpose of low-level physics tests) shall the temperature of the reactor vessel be less than the minimum permissible temperature for the inservice system hydrostatic pressure test nor less than 40°F above that temperature required by section IV.A.2.a.

d. If there is no fuel in the reactor during the initial preoperational system leakage and hydrostatic pressure tests, the minimum permissible test temperature shall be determined in accordance with paragraph G2410 of the ASME Code except that the factor of safety applied to each term making up the calculated stress intensity factor may be reduced to 1.0. In no case shall the test temperature be less than $RT_{NDT}+60°F$.

3. Materials for piping (i.e., pipe, tubes and fittings), pumps, and valves (excluding bolting materials) shall meet the requirements of paragraph G3100 of the ASME Code.

4. Materials for bolting and other fasteners with nominal diameters exceeding 1 inch shall meet the minimum requirements of 25 mils lateral expansion and 45 ft lbs in terms of Charpy V-notch tests conducted at the preload temperature or at the lowest service temperature, whichever temperature is lower.

B. Reactor vessels beltline materials shall have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirradiated specimens in accordance with paragraphs NB-2322.2(4) and 2322.2(6) of the ASME Code, of 75 ft lbs unless it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper shelf fracture energy are adequate.

C. Reactor vessels for which the predicted value of adjusted reference temperature exceeds 200°F shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. INSERVICE REQUIREMENTS—REACTOR VESSEL BELTLINE MATERIAL

A. The properties of reactor vessel beltline region materials, including welds, shall be monitored by a material surveillance program conforming to the "Reactor Vessel Material Surveillance Program Requirements" set forth in Appendix H.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of section IV.A.2. are satisfied, using the predicted value of the adjusted reference temperature at the end of the service period to account for the effects of irradiation on the fracture toughness of the beltline materials. The basis for the prediction shall include results from pertinent radiation effects studies in addition to the results of the surveillance program of section V.A.

C. In the event that the requirements of section V.B. cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

1. An esentially complete volumetric examination of the beltline region of the vessel including 100 percent of any weldments shall be made in accordance with the requirements of Section XI of the ASME Code.

2. Additional evidence of the changes in fracture toughness of the beltline materials resulting from exposure to neutron irradiation shall be obtained from results of supplemental tests, such as measurements of dynamic fracture toughness of archive material that has been subjected to accelerated irradiation.

3. A fracture analysis shall be performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of adequate margins for continued operation.

D. If the procedures of section V.C. do not indicate the existence of an adequate safety margin, the reactor vessel beltline region shall be subjected to a thermal annealing treatment to effect recovery of material toughness properties. The degree of such recovery shall be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and annealed under the same time-at-temperature conditions as those given the beltline material. The results shall provide the basis for establishment of the adjusted reference temperature after annealing. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of section IV.A.2., using the values of adjusted reference temperature that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of sections V.C. and V.D. shall be reported to the Commission for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of section V.B.