Summary of the basis for the 10 CFR 50 Appendix G 50 ft-lb Criteria

Draft of 21 April 2011

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Note: *The summary information is a draft document and therefore cannot be cited in its current form. It is provided for your information only. However, all of the documents referred to in the summary are publically available, and are included in this PDF file. These documents may be cited.*

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Fracture Toughness Requirements for Light-Water Nuclear Power Reactors

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Commission is amending its regulations which specify fracture toughness requirements for light-water nuclear power reactors and its requirements for reactor vessel material surveillance programs. The amendments clarify the applicability of these requirements to all plants, modify certain requirements, and shorten and simplify these regulations by more extensively incorporating by reference appropriate National Standards.

EFFECTIVE DATE: July 26, 1983.

FOR FURTHER INFORMATION CONTACT: Dr. P. N. Randall, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone (301) 443-5903.

SUPPLEMENTARY INFORMATION: On November 14, 1980 the Nuclear Regulatory Commission published in the **Federal Register** (45 FR 75536) proposed amendments to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which would amend Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements." These amendments comprised a proposed general revision of Appendices G and H designed to update them after seven years of use and to make them more consistent with current technology and pertinent National Standards. Interested persons were invited to submit written comment by January 13, 1981. Thirteen letters of comment were received. All were from utilities or vendors and addressed the application of specific requirements contained in the proposed rule. There were no adverse general comments or objections to the proposed revisions. A brief summary of the more significant comments and the staff responses follows:

The most significant technical question, which affects pressure-temperature limits for all plants, concerned a new requirement for fracture control at structural discontinuities, contained in paragraph IV.A.2 of Appendix G. The critical locations are the closure flange regions of the reactor vessel where bending stress is introduced during boltup. The requirement in the proposed rule was that the temperature at the highly stressed region be at least 150°F above the reference temperature of the material whenever pressure exceeded 20 percent of the preoperational system hydrostatic test pressure. Commenters felt this was overly restrictive, and cited certain hardships caused during hydrotests and normal heatup and cooldown operations. In response to the comments, the requirement has been revised to provide a separate, lower temperature requirement for hydrotest conditions than for normal operation, consistent with the margins of safety specified in the ASME Code. In addition, the temperature requirement for normal operation was reduced slightly based on further analysis of boltup conditions. Thus, in the final rule, the proposed requirement of 150°F (above the reference temperature of the material) was revised to 90°F for hydrotest and 120°F for normal operation. This requirement will affect principally those plants where radiation damage to the beltline region is low, and the pressure-temperature limits are thus more likely to be controlled by the closure flange regions.

Paragraph IV.A.4. of Appendix G was expanded to specify that the quantity "RT_{NDT}+60°F" referred to the adjusted reference temperature of the reactor vessel material in the region that was controlling the pressure-temperature limits (beltline or closure flange regions) following the analysis required by paragraph IV.A.2.

The requirements concerning thermal annealing of reactor vessels, given in paragraphs IV.B. and V.D. of Appendix G, represent no basic change from those published in 1973. However, the recent investigation of pressurized thermal shock effects prompted some studies of annealing to identify and resolve possible engineering difficulties. If the results show that changes should be made in paragraph IV.B. or V.D., a further amendment to Appendix G will be issued.

Minor changes in wording were made in several paragraphs, and footnotes were added to clarify the meaning of two paragraphs.

A number of comments addressed the reporting requirements for surveillance reports, paragraphs III.A. and III.C. of Appendix H. Based on commenters suggestions, the Commission has revised the proposed requirement that surveillance reports be submitted within 90 days after completion of testing to require submittal of these reports within 1 year of capsule withdrawal unless an extension is granted. This change simplifies implementation of the requirement, because capsule withdrawal schedules must be approved by the Director, Office of Nuclear Reactor Regulation, as provided in paragraph II.B.3. of Appendix H. The primary purposes of the requirement—timely reporting of test results and notification of any problems—are accomplished as well by the provisions of the final rule.

Copies of the abstract of comments and the staff's response, which gives a point-by-point discussion of each issue raised by the commenters, and copies of the value-impact analysis supporting the rule are available for public inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, DC. Single copies may be obtained by written request to the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: P. N. Randall.

Regulatory Analysis

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW., Washington, DC. Single copies of the analysis may be obtained from P. N. Randall, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone (301) 443-5903.

Paperwork Reduction Act Statement

The reporting and recordkeeping requirements contained in this regulation have been approved by the Office of Management and Budget, OMB approval No. 3150-0011.

Regulatory Flexibility Statement

In accordance with Section 605(b) of the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not have a significant economic impact on a substantial number of small entities. This rule affects primarily the utilities that own light water nuclear power reactors, and the vendors of those reactors, none of which meet the definition of "small entities" set forth in Section 601(3) of the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration in 13 CFR Part 121.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, the following amendments to 10 CFR Part 50 are published as a document subject to codification.

1. The authority citation for Part 50 is revised to read as follows:

Authority: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.58, 50.91 and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10 (a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10 (b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. Paragraph (a) of § 50.12 is revised to read as follows:

§ 50.12 Specific exemptions.

(a) The Commission may, upon application by any interested person or upon its own initiative grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the pubic interest. To obtain an exemption to Appendices G and H to this part, the requirements of paragraph 50.60(b) of this part must be met in addition to the requirements of this paragraph.

* * * * *

§ 50.55a [Amended]

3. In § 50.55a, paragraph (i) is removed and paragraph (j) is redesignated paragraph (i).

4. A new § 50.60 is added to 10 CFR Part 50 to read as follows:

§ 50.60 Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12. In addition, the applicant must demonstrate that (i) compliance with the specified requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety, and (ii) the proposed alternatives would provide an adequate level of quality and safety.

5. Appendices G and H are revised to read as follows:

Appendix G—Fracture Toughness Requirements

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I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear 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 ASME Code forms the basis for the requirements of this Appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified the applicable ASME Code edition and addenda and any limitations and modifications thereof are specified in § 50.55a of this part.

The ASME Boiler and Pressure Vessel Code has been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the **Federal Register**. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

The requirements of this appendix apply to the following materials:

Note.—The adequacy of the fracture toughness of other ferritic materials not covered in this section shall be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G-2110 of the ASME Code as defined in paragraph II.A. of this appendix. The latest edition and addenda permitted by paragraph 50.55a(b) of this part at the time the analysis is made is to be used for the purpose of this paragraph.

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

II. Definitions

A. "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 crystal structure.

B. "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.

C. "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 under § 50.55a of this part.

D. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in the ASME Code.

E. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Section V of this Appendix) by adding to RT_{NDT} the temperature shift, measured at the 30 ft-lb (41J) level, in the average Charpy curve for the irradiated material relative to that for the unirradiated material.

F. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Sections IV and V of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under § 50.55a of this part), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this Appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph V.C.2. of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs A and B of this section must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

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

1. Reactor vessel beltline materials must have Charpy upper-shelf energy [1] of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The latest edition and addenda of the ASME Code permitted by paragraph 50.55a(b) of this part at the time the analysis is made are to be used for the purposes of paragraphs IV.A.1 and IV.A.2 of this appendix.

2. When the core is not critical, pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code supplemented by the requirements of Section V of this appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F (67°C) for normal operation and by 90°F (50°C) for hydrostatic pressure tests and leak tests, unless a lower temperature can be justified by showing that the margins of safety for those regions when they are controlling are equivalent to those required for the beltline when it is controlling. The justification submitted for the pressure temperature limits must describe the methods of analysis used.

3. When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F (22°C) above the minimum permissible temperature of paragraph 2. of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

4. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature must be 60°F (33°C) above the adjusted reference temperature of the reactor vessel material in the region that is controlling (as specified in paragraph IV.A.2 of this appendix).

5. If there is fuel in the reactor during system hydrostatic pressure tests or leak tests, the requirements of paragraphs 2 or 3 of this section apply, depending on whether the core is critical during the test.

B. Reactor vessels for which the predicted value of upper-shelf energy at end of life is below 50 ft-lb or for which the predicted value of adjusted reference temperature at end of life exceeds 200°F (93°C) must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. Inservice Requirements—Reactor Vessel Beltline Material

A. The effects of neutron radiation on the reference temperature and upper shelf energy of reactor vessel beltline materials, including welds, are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H to this part.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV of this appendix are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions are to be made for the radiation conditions at the critical location on the crack front of the assumed flaw.[2] The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials must be used to verify that the fracture toughness requirements are satisfied.

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

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B. of this appendix is made and any flaws characterized according to Section XI of the ASME Code and as otherwise specified by the Director, Office of Nucelar Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation is to be obtained from results of supplemental fracture toughness tests.

3. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

D. If the procedures of Section V.C. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may, subject to the approval of the Director, Office of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material. The degree of recovery is to be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-at-temperature conditions as those given the beltline material. The results, together with the results of other pertinent annealing-effects studies, are to provide the basis for establishing the adjusted reference temperature and upper-shelf energy 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. of this appendix using the values of adjusted reference temperature and upper-shelf energy that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of Sections V.C. and V.D. of this appendix are to be reported to the Director, Office of Nuclear Reactor Regulation, as specified in § 50.4(a) of this Part, 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 of this appendix.

[1] Defined in ASTM E 185–79 and –82 which are incorporated by reference in Appendix H.

[2] For example, in analyses that follow Appendix G of the ASME Code, the radiation conditions to be used are those predicted for the material one fourth of the way through the vessel wall, i.e., at the deepest point on the crack front of the postulated defect.

Appendix H—Reactor Vessel Material Surveillance Program Requirements

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I. Introduction

The purpose of the material surveillance program required by this Appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Sections IV and V of Appendix G to this part.

ASTM E 185–73, –79 and –82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the **Federal Register.** Copies of ASTM E 185–73, –79, and –82, may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, PA 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

II. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E>1MEV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

B. Reactor vessels that do not meet the conditions of paragraph II.A. of this Appendix must have their beltline materials monitored by this Appendix.

1. That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185–82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the ASME Code. The design and location of the capsule holders shall permit insertion of replacement capsules. Accelerated irradiation capsule may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

3. A proposed withdrawal schedule must be submitted with a technical justification therefor to the Director, Office of Nuclear Reactor Regulation, for approval. The proposed schedule must be approved prior to implementation.

C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations:

1. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

2. There must be adequate arrangement for data sharing between plants.

3. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

4. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

III. Report of Test Results

A. Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted to the Director, Office of Nuclear Reactor Regulation, as specified in § 50.4(a) of this Part, within 1 year after capsule withdrawal, unless an extension is granted by the Director.

B. The report must include the data required by ASTM E 185, as specified in paragraph II.B.1 of this Appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specifications must be provided with the report.

Dated at Washington, D.C. this 23d day of May 1983.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

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at the Federal repository. All of these high-level radioactive wastes shall be transferred to a Federal repository no later than 10 years following separation of fission products from the irradiated fuel. Upon receipt, the Federal repository will assume permanent custody of these radioactive waste materials although industry will pay the Federal Government a charge which together with interest on unexpended balances will be designed to defray all costs of disposal and perpetual surveillance, the Department of Energy will take title to the radioactive waste material upon transfer to a Federal repository. Before retirement of the reprocessing plant from operational status and before termination of licensing pursuant to § 50.82, transfer of all such wastes to a Federal repository shall be completed. Federal repositories, which will be limited in number, will be designated later by the Commission.

3. Disposal of high-level radioactive fission product waste material will not be permitted on any land other than that owned and controlled by the Federal Government.

4. A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned. Criteria for the extent of decontamination to be required upon decommissioning and license termination will be developed in consultation with competent groups. Opportunity will be afforded for public comment before such criteria are made effective.

5. Applicants proposing to operate fuel reprocessing plants, in submitting information concerning financial qualifications as required by § 50.33(f), shall include information enabling the Commission to determine whether the applicant is financially qualified, among other things, to provide for the removal and disposal of radioactive wastes, during operation and upon decommissioning of the facility, in accordance with the Commission's regulations, including the requirements set out in this appendix.

6. With respect to fuel reprocessing plants already licensed, the licenses will be appropriately conditioned to carry out the purposes of the policy stated above with respect to high-level radioactive fission product wastes generated after installation of new equipment for interim storage of liquid wastes, or after installation of equipment required for solidification without interim liquid storage. In either case, such equipment shall be installed at the earliest practicable date, taking into account the time required for design, procurement and installation thereof. With respect to such plants, the application of the policy stated in this appendix to existing wastes and to wastes generated prior to the installation of such equipment, will be the subject of a further rule making proceeding.

(42 U.S.C. 2201, 2237; sec. 161, Pub. L. 83-703; 68 Stat. 948 (42 U.S.C. 2201); sec. 201, Pub. L. 93-438, 88 Stat. 1242, (42 U.S.C. 5841))

[35 FR 17533, Nov. 14, 1970, as amended at 36 FR 5411, Mar. 23, 1971; 42 FR 20139, Apr. 18, 1977; 45 FR 14201, Mar. 5, 1980]

Appendix G—Fracture Toughness Requirements

Table of Contents

I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear 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 ASME Code forms the basis for the requirements of this appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified the applicable ASME Code edition and addenda and any limitations and modifications thereof are specified in § 50.55a of this part.

The ASME Boiler and Pressure Vessel Code has been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

The requirements of this appendix apply to the following materials:

NOTE: The adequacy of the fracture toughness of other ferritic materials not covered in this section shall be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G-2110 of the ASME Code as defined in paragraph II.A. of this appendix. The latest edition and addenda permitted by paragraph 50.55a(b) of this part at the time the analysis is made is to be used for the purpose of this paragraph.

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

II. DEFINITIONS

A. "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 crystal structure.

B. "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.

C. "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 under § 50.55a of this part.

D. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in the ASME Code.

E. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Section V of this appendix) by adding to RT_{NDT} the temperature shift, measured at the 30 ft-lb (41J) level, in the average Charpy curve for the irradiated material relative to that for the unirradiated material.

F. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. FRACTURE TOUGHNESS TESTS

A. To demonstrate compliance with the fracture toughness requirements of Sections IV and V of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under § 50.55a of this part), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph V.C.2. of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs A and B of this section must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. FRACTURE TOUGHNESS REQUIREMENTS

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

1. Reactor vessel beltline materials must have Charpy upper-shelf energy [1] of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The latest edition and addenda of the ASME Code permitted by paragraph 50.55a(b) of this part at the time the analysis is made are to be used for the purposes of paragraphs IV.A.1 and IV.A.2 of this appendix.

[1] Defined in ASTM E 185-79 and -82 which are incorporated by reference in Appendix H.

2. When the core is not critical, pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code supplemented by the requirements of Section V of this appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F (67°C) for normal operation and by 90°F (50°C) for hydrostatic pressure tests and leak tests, unless a lower temperature can be justified by showing that the margins of safety for those regions when they are controlling are equivalent to those required for the beltline when it is controlling. The justification submitted for the pressure temperature limits must describe the methods of analysis used.

3. When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F (22°C) above the minimum permissible temperature of paragraph 2. of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

4. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature must be 60°F (33°C) above the adjusted reference temperature of the reactor vessel material in the region that is controlling (as specified in paragraph IV.A.2 of this appendix).

5. If there is fuel in the reactor during system hydrostatic pressure tests or leak tests, the requirements of paragraphs 2 or 3 of this section apply, depending on whether the core is critical during the test.

B. Reactor vessels for which the predicted value of upper-shelf energy at end of life is below 50 ft-lb or for which the predicted value of adjusted reference temperature at end of life exceeds 200°F (93°C) must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. INSERVICE REQUIREMENTS—REACTOR VESSEL BELTLINE MATERIAL

A. The effects of neutron radiation on the reference temperature and upper shelf energy of reactor vessel beltline materials, including welds, are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H to 'this part.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV of this appendix are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions are to be made for the radiation conditions at the critical location on the crack front of the assumed flaw.[2] The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials must be used to verify that the fracture toughness requirements are satisfied.

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

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B. of this appendix is made and any flaws characterized according to Section XI of the ASME Code and as otherwise specified by the Director, Office of Nuclear Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation is to be obtained from results of supplemental fracture toughness tests.

3. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

D. If the procedures of Section V.C. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may, subject to the approval of the Director, Office of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material. The degree of recovery is to

[2] For example, in analyses that follow Appendix G of the ASME Code, the radiation conditions to be used are those predicted for the material one fourth of the way through the vessel wall, i.e., at the deepest point on the crack front of the postulated defect.

be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-at-temperature conditions as those given the beltline material. The results, together with the results of other pertinent annealing-effects studies, are to provide the basis for establishing the adjusted reference temperature and upper-shelf energy 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. of this appendix using the values of adjusted reference temperature and upper-shelf energy that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of Sections V.C. and V.D. of this appendix are to be reported to the Director, Office of Nuclear Reactor Regulation, as specified in § 50.4(a) of this part, 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 of this appendix.

[48 FR 24009, May 27, 1983]

Appendix H—Reactor Vessel Material Surveillance Program Requirements

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I. Introduction

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Sections IV and V of Appendix G to this part.

ASTM E 185-73, -79 and -82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of ASTM E 185-73, -79, and -82, may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, PA 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

II. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E>IMEV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2

B. Reactor vessels that do not meet the conditions of paragraph II.A. of this Appendix must have their beltline materials monitored by this appendix.

1. That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the ASME Code. The design and location of the capsule holders shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

3. A proposed withdrawal schedule must be submitted with a technical justification therefor to the Director, Office of Nuclear Reactor Regulation, for approval. The pro-

posed schedule must be approved prior to implementation.

C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations:

1. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

2. There must be adequate arrangement for data sharing between plants.

3. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

4. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

III. REPORT OF TEST RESULTS

A. Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted to the Director, Office of Nuclear Reactor Regulation, as specified in § 50.4(a) of this Part, within 1 year after capsule withdrawal, unless an extension is granted by the Director.

B. The report must include the data required by ASTM E 185, as specified in paragraph II.B.1 of this appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specifications must be provided with the report.

[48 FR 24011, May 27, 1983]

APPENDIX I—NUMERICAL GUIDES FOR DESIGN OBJECTIVES AND LIMITING CONDITIONS FOR OPERATION TO MEET THE CRITERION "AS LOW AS IS REASONABLY ACHIEVABLE" FOR RADIOACTIVE MATERIAL IN LIGHT-WATER-COOLED NUCLEAR POWER REACTOR EFFLUENTS

SECTION I. *Introduction.* Section 50.34a provides that an application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application must also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as practicable.

Section 50.36a contains provisions designed to assure that releases of radioactive material from nuclear power reactors to unrestricted areas during normal reactor operations, including expected operational occurrences, are kept as low as practicable.

This appendix provides numerical guides for design objectives and limiting conditions for operation to assist applicants for, and holders of, licenses for light-water-cooled nuclear power reactors in meeting the requirements of §§ 50.34a and 50.36a that radioactive material in effluents released from these facilities to unrestricted areas be kept as low as is reasonably achievable. Design objectives and limiting conditions for operation conforming to the guidelines of this appendix shall be deemed a conclusive showing of compliance with the "as low as is reasonably achievable" requirements of 10 CFR 50.34a and 50.36a. Design objectives and limiting conditions for operation differing from the gnidelines may also be used, subject to a case-by-case showing of a sufficient basis for the findings of "as low as is reasonably achievable" required by §§ 50.34a and 50.36a. The guides presented in this appendix are appropriate only for light-water-cooled nuclear power reactors and not for other types of nuclear facilities.

SEC. II. *Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.* The guides on design objectives set forth in this section may be used by an applicant for a permit to construct a light-water-cooled nuclear power reactor as guidance in meeting the requirements of § 50.34a(a). The applicant shall provide reasonable assurance that the following design objectives will be met.

A. The calculated annual total quantity of all radioactive material above background[1] to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

2. Notwithstanding the guidance of paragraph B.1:

(a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

C. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

[1]Here and elsewhere in this appendix background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors of which specific account is required in determining design objectives.

D. In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis. The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25-30, reproduced in the Annex to this Appendix I.

SEC. III. *Implementation.* A.1. Conformity with the gnides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimations of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation: *Provided,* That, if the requirements of paragraph B of Section III are fulfilled, the applicant shall be deemed to have complied with the requirements of paragraph C of Section II with respect to radioactive iodine if estimations of exposure are made on the basis of such food pathways and individual receptors as actually exist at the time the plant is licensed.

2. The characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of dis-

charge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

B. If the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values of paragraph C of Section II, the applicant shall provide reasonable assurance that a monitoring and surveillance program will be performed to determine:

1. The quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives;

2. Whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and

3. The content of radioactive iodine and foods involved in the changes, if and when they occur.

SEC. IV. *Guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.* The guides on limiting conditions for operation for light-water-cooled nuclear power reactors set forth below may be used by an applicant for a license to operate a light-water-cooled nuclear power reactor as guidance in developing technical specifications under § 50.36a(a) to keep levels of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable.

Section 50.36a(b) provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications that take into account the need for operating flexibility and at the same time assure that the licensee will exert his best effort to keep levels of radioactive material in effluents as low as is reasonably achievable. The guidance set forth below provides additional and more specific guidance to licensees in this respect.

Through the use of the guides set forth in this Section it is expected that the annual releases of radioactive material in effluents from light-water-cooled nuclear power reactors can generally be maintained within the levels set forth as numerical guides for design objectives in Section II.

At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such numerical guides for design objectives but still within levels that assure that the average population exposure is equivalent to small fractions of doses from natural background radiation. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents within the numerical guides for design objectives.

A. If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water-cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall: [2]

1. Make an investigation to identify the causes for such release rates;

2. Define and initiate a program of corrective action; and

3. Report these actions to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this Chapter with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days from the end of the quarter during which the release occurred.

B. The licensee shall establish an appropriate surveillance and monitoring program to:

1. Provide data on quantities of radioactive material released in liquid and gaseous effluents to assure that the provisions of paragraph A of this section are met;

2. Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure; and

3. Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.

C. If the data developed in the surveillance and monitoring program described in paragraph B of this section and in para-

[2] Section 50.36a(a)(2) requires the licensee to submit certain reports to the Commission with regard to the quantities of the principal radionuclides released to unrestricted areas. It also provides that, on the basis of such reports and any additional information the Commission may obtain from the licensee and others, the Commission may from time to time require the license to take such action as the Commission deems appropriate.

graph B of Section III or from other monitoring programs show that the relationship between the quantities of radioactive material released in liquid and gaseous effluents and the dose to individuals in unrestricted areas is significantly different from that assumed in the calculations used to determine design objectives pursuant to Sections II and III, the Commission may modify the quantities in the technical specifications defining the limiting conditions for operation in a license authorizing operation of a light-water-cooled nuclear power reactor.

SEC. V. *Effective dates.* A. The guides for limiting conditions for operation set forth in this appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a permit to construct a light-water-cooled nuclear power reactor.

B. For each light-water-cooled nuclear power reactor constructed pursuant to a permit for which application was filed prior to January 2, 1971, the holder of the permit or a license, authorizing operation of the reactor shall, within a period of twelve months from June 4, 1975, file with the Commission:

1. Such information as is necessary to evaluate the means employed for keeping levels of radioactivity in effluents to unrestricted areas as low as is reasonably achievable, including all such information as is required by § 50.34a (b) and (c) not already contained in his application; and

2. Plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable.

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF (DOCKET-RM-50-2)

GUIDES ON DESIGN OBJECTIVES FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

A. For radioactive material above background [1] in liquid effluents to be released to unrestricted areas:

1. The calculated annual total quantity of all radioactive material from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 5 millirems; and

2. The calculated annual total quantity of radioactive material, except tritium and dissolved gases, should not exceed 5 curies for each light-water-cooled reactor at a site.

3. Notwithstanding the guidance in paragraph A.2, for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures [2] to reduce the possible sources of radioactive material in liquid effluent releases and the calculated quantity exceeds the quantity set forth in paragraph A.2, the requirements for design objectives for radioactive material in liquid effluents may be deemed to have been met provided:

a. The applicant submits an evaluation of the potential for effects from long-term buildup in the environment in the vicinity of the site of radioactive material, with a radioactive half-life greater than one year, to be released; and

b. The provisions of paragraph A.1 are met.

B. For radioactive material above background in gaseous effluents the annual total quantity of radioactive material to be released to the atmosphere by all light-water-cooled nuclear power reactors at a site:

1. The calculated annual air dose due to gamma radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 10 millirads; and

2. The calculated annual air dose due to beta radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 20 millirads.

3. Notwithstanding the guidance in paragraphs B.1 and B.2, for a particular site:

a. The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background in gaseous effluents to be released to the atmosphere if it appears that the use of the design objectives described in paragraphs B.1 and B.2 is likely to result in an annual dose to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin; or

b. Design objectives based on a higher quantity of radioactive material above background in gaseous effluents to be released to the atmosphere than the quantity specified in paragraphs B.1 and B.2 may be deemed

[1] "Background," means the quantity of radioactive material in the effluent from light-water-cooled nuclear power reactors at a site that did not originate in the reactors.

[2] Such measures may include treatment of clear liquid waste streams (normally tritiated, nonaerated, low conductivity equipment drains and pump seal leakoff), dirty liquid waste streams (normally nontritiated, aerated, high conductivity building sumps, floor and sample station drains), steam generator blowdown streams, chemical waste streams, low purity and high purity liquid streams (resin regenerate and laboratory wastes), as appropriate for the type of reactor.

to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as practicable if the applicant provides reasonable assurance that the proposed higher quantity will not result in annual doses to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin.

C. For radioactive iodine and radioactive material in particulate form above background released to the atmosphere:

1. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 millirems. In determining the dose or dose commitment the portion thereof due to intake of radioactive material via the food pathways may be evaluated at the locations where the food pathways actually exist; and

2. The calculated annual total quantity of iodine-131 in gaseous effluents should not exceed 1 curie for each light-water-cooled nuclear power reactor at a site.

3. Notwithstanding the guidance in paragraphs C.1 and C.2 for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures [3] to reduce the possible sources of radioactive iodine releases, and the calculated annual quantities taking into account such control measures exceed the design objective quantities set forth in paragraphs C.1 and C.2, the requirements for design objectives for radioactive iodine and radioactive material in particulate form in gaseous effluents may be deemed to have been met provided the calculated annual total quantity of all radioactive iodine and radioactive material in particulate form that may be released in gaseous effluents does not exceed four times the quantity calculated pursuant to paragraph C.1.

[40 FR 19442, May 5, 1975, as amended at 40 FR 40818, Sept. 4, 1975; 40 FR 58847, Dec. 19, 1975; 41 FR 16447, Apr. 19, 1976; 42 FR 20139, Apr. 18, 1977]

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[3] Such in-plant control measures may include treatment of steam generator blowdown tank exhaust, clean steam supplies for turbine gland seals, condenser vacuum systems, containment purging exhaust and ventilation exhaust systems and special design features to reduce contaminated steam and liquid leakage from valves and other sources such as sumps and tanks, as appropriate for the type of reactor.

APPENDIX J—PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS

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I. INTRODUCTION

One of the conditions of all operating licenses for water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for such tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

II. EXPLANATION OF TERMS

A. "Primary reactor containment" means the structure or vessel that encloses the components of the reactor coolant pressure boundary, as defined in § 50.2(v), and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

B. "Containment isolation valve" means any valve which is relied upon to perform a containment isolation function.

C. "Reactor containment leakage test program" includes the performance of Type A, Type B, and Type C tests, described in II.F, II.G, and II.H, respectively.

D. "Leakage rate" for test purposes is that leakage which occurs in a unit of time, stated as a percentage of weight of the original content of containment air at the leakage rate test pressure that escapes to the outside atmosphere during a 24-hour test period.

E. "Overall integrated leakage rate" means that leakage rate which obtains from a summation of leakage through all potential leakage paths including containment welds, valves, fittings, and components which penetrate containment.

F. "Type A Tests" means tests intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter.

G. "Type B Tests" means tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:

1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant componds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.

3. Doors with resilient seals or gaskets except for seal-welded doors.

4. Components other than those listed in II.G.1, II.G.2, or II.G.3 which must meet the acceptance criteria in III.B.3.

H. "Type C Tests" means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under postaccident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

I. Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases.

J. Pt (p.s.i.g.) means the containment vessel reduced test pressure selected to measure the integrated leakage rate during periodic Type A tests.

K. La (percent/24 hours) means the maximum allowable leakage rate at pressure Pa as specified for preoperational tests in the technical specifications or associated bases, and as specified for periodic tests in the operating license.

L. Ld (percent/24 hours) means the design leakage rate at pressure, Pa, as specified in the technical specifications or associated bases.

M. Lt (percent/24 hours) means the maximum allowable leakage rate at pressure Pt derived from the preoperational test data as specified in III.A.4.(a)(iii).

N. Lam, Ltm (percent/24 hours) means the total measured containment leakage rates at pressure Pa and Pt, respectively, obtained from testing the containment with components and systems in the state as close as practical to that which would exist under design basis accident conditions (e.g., vented, drained, flooded or pressurized).

O. "Acceptance criteria" means the standard against which test results are to be compared for establishing the functional acceptability of the containment as a leakage limiting boundary.

III. Leakage Testing Requirements

A program consisting of a schedule for conducting Type A, B, and C tests shall be developed for leak testing the primary reactor containment and related systems and components penetrating primary containment pressure boundary.

Upon completion of construction of the primary reactor containment, including installation of all portions of mechanical, fluid, electrical, and instrumentation systems penetrating the primary reactor containment pressure boundary, and prior to any reactor operating period, preoperational and periodic leakage rate tests, as applicable, shall be conducted in accordance with the following:

A. *Type A test*—1. *Pretest requirements.* (a) Containment inspection in accordance with V.A. shall be performed as a prerequisite to the performance of Type A tests. During the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested in as close to the "as is" condition as practical. During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments shall be made to components whose leakage exceeds that specified in the technical specification as soon as practical after identification. If during a Type A test, including the supplemental test specified in III.A.3.(b), potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria III.A.4.(b) or III.A.5.(b), the Type A test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and a Type A test performed. The corrective action taken and the change in leakage rate determined from the tests and overall integrated leakage determined from the local leak and Type A tests shall be included in the report submitted to the Commission as specified in V.B.

(b) Closure of containment isolation valves for the Type A test shall be accomplished by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor). Repairs of maloperating or leaking valves shall be made as necessary. Information on any valve closure malfunction or valve leakage that requires corrective action before the test, shall be included in the report submitted to the Commission as specified in V.B.

(c) The containment test conditions shall stabilize for a period of about 4 hours prior to the start of a leakage rate test.

(d) Those portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment shall be opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident shall be vented to the containment atmosphere. All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure and to assure they will be subjected to the post-accident differential pressure. Systems that are required to maintain the plant in a safe condition during the test shall be operable in their normal mode, and need not be vented. Systems that are normally filled with water and operating under post-accident conditions, such as the containment heat removal system, need not be vented. However, the containment isolation valves in the systems defined in III.A.1.(d) shall be tested in accordance with III.C. The measured leakage rate from these tests shall be reported to the Commission.

2. *Conduct of tests.* Preoperational leakage rate tests at either reduced or at peak pressure, shall be conducted at the intervals specified in III.D.

3. *Test methods.* (a) All Type A tests shall be conducted in accordance with the provisions of the American National Standard N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors, March 16, 1972.[1] The method chosen for the initial test shall normally be used for the periodic tests.

(b) The accuracy of any Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4-1972. The supplemental test method selected shall be conducted for sufficient duration to establish accurately the change in leakage rate between the Type A and supplemental test. Results from this supplemental test are acceptable provided the difference between the supplemental test data and the Type A test data is within 0.25 La (or 0.25 Lt). If results are not within 0.25 La (or 0.25 Lt), the reason shall be determined, corrective action taken, and a successful supplemental test performed.

(c) Test leakage rates shall be calculated using absolute values corrected for instrument error.

4. *Preoperational leakage rate tests.* (a) *Test pressure*—(1) *Reduced pressure tests.* (i) An initial test shall be performed at a pressure Pt, not less than 0.50 Pa to measure a leakage rate Ltm.

(ii) A second test shall be performed at pressure Pa to measure a leakage rate Lam.

(iii) The leakage characteristics yielded by measurements Ltm and Lam shall establish the maximum allowable test leakage rate Lt of not more than La (Ltm/Lam). In the event Ltm/Lam is greater than 0.7, Lt shall be specified as equal to La (Pt/Pa).[1, 2]

(2) *Peak pressure tests.* A test shall be performed at pressure Pa to measure the leakage rate Lam.

(b) *Acceptance criteria*—(1) *Reduced pressure tests.* The leakage rate Ltm shall be less than 0.75 Lt.

(2) *Peak pressure tests.* The leakage rate Lam shall be less than 0.75 La and not greater than Ld.

5. *Periodic leakage rate tests*—(a) *Test pressure.* (1) Reduced pressure tests shall be conducted at Pt;

(2) Peak pressure tests shall be conducted at Pa.

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[1] ANSI N45.4-1972 Leakage Rate Testing of Containment Structures for Nuclear Reactors (dated Mar. 16, 1972). Copies may be obtained from the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, IL 60521. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, DC. The incorporation by reference was approved by the Director of the Federal Register on October 30, 1972.

[2] Such inservice inspections are required by § 50.55a.

(b) *Acceptance criteria*—(1) *Reduced pressure tests.* The leakage rate Ltm shall be less than 0.75 Lt. If local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements shall be taken at a test pressure Pt.

(2) *Peak pressure tests.* The leakage rate Lam shall be less than 0.75 La. If local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements shall be taken at a test pressure Pa.

6. *Additional requirements.* (a) If any periodic Type A test fails to meet the applicable acceptance criteria in III.A.5.(b), the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission.

(b) If two consecutive periodic Type A tests fail to meet the applicable acceptance criteria in III.A.5(b), notwithstanding the periodic retest schedule of III.D., a Type A test shall be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria in III.A.5(b), after which time the retest schedule specified in III.D. may be resumed.

B. *Type B tests*—1. *Test methods.* Acceptable means of performing preoperation and periodic Type B tests include:

(a) Examination by halide leak-detection method (or by other equivalent test methods such as mass spectrometer) of a test chamber, pressurized with air, nitrogen, or pneumatic fluid specified in the technical specifications or associated bases and constructed as part of individual containment penetrations.

(b) Measurement of the rate of pressure loss of the test chamber of the containment penetration pressurized with air, nitrogen, or pneumatic fluid specified in the technical specifications or associated bases.

(c) Leakage surveillance by means of a permanently installed system with provisions for continuous or intermittent pressurization of individual or groups of containment penetrations and measurement of rate of pressure loss of air, nitrogen, or pneumatic fluid specified in the technical specification or associated bases through the leak paths.

2. *Test pressure.* All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.

3. *Acceptance criteria.* (See also Type C tests.) (a) The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves specified in III.C.3.

(b) Leakage measurements obtained through component leakage surveillance systems (e.g., continuous pressurization of individual containment components) that maintains a pressure not less than Pa at individual test chambers of containment penetrations during normal reactor operation, are acceptable in lieu of Type B tests.

C. *Type C tests*—1. *Test method.* Type C tests shall be performed by local pressurization. The pressure shall be applied in the same direction as that when the value would be required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. The test methods in III.B.1 may be substituted where appropriate. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor).

2. *Test pressure.* (a) Valves, unless pressurized with fluid (e.g., water, nitrogen) from a seal system, shall be pressurized with air or nitrogen at a pressure of Pa.

(b) Valves, which are sealed with fluid from a seal system shall be pressurized with that fluid to a pressure not less than 1.10 Pa.

3. *Acceptance criterion.* The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La. Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate: *Provided,* That;

(a) Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the technical specifications or associated bases, and

(b) The installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days at a pressure of 1.10 Pa.

D. *Periodic retest schedule*—1. *Type A test.* (a) After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections. [2]

(b) Permissible periods for testing. The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under the administrative control and in accordance with the safety procedures defined in the license.

2. *Type B tests.* (a) Type B tests, except tests for air locks, shall be performed during

[2] Such inservice inspections are required by § 50.55a.

reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than 2 years. If opened following a Type A or B test, containment penetrations subject to Type B testing shall be Type B tested prior to returning the reactor to an operating mode requiring containment integrity. For primary reactor containment penetrations employing a continuous leakage monitoring system, Type B tests, except for tests of air locks, may, notwithstanding the test schedule specified under III.D.1., be performed every other reactor shutdown for refueling but in no case at intervals greater than 3 years.

(b)(i) Air locks shall be tested prior to initial fuel loading and at 6-month intervals thereafter at an internal pressure not less than P_a.

(ii) Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a.

(iii) Air locks opened during periods when containment integrity is required by the plant's Technical Specifications shall be tested within 3 days after being opened. For air lock doors opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings. For air lock doors having testable seals, testing the seals fulfills the 3-day test requirements. In the event that the testing for this 3-day interval cannot be at P_a, the test pressure shall be as stated in the Technical Specifications. Air lock door seal testing shall not be substituted for the 6-month test of the entire air lock at not less than P_a.

(iv) The acceptance criteria for air lock testing shall be stated in the Technical Specifications.

3. *Type C tests.* Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years.

IV. Special Testing Requirements

A. *Containment modification.* Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable for the area affected by the modification. The measured leakage from this test shall be included in the report to the Commission, required by V.A. The acceptance criteria of III.A.5.(b), III.B.3., or III.C.3., as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

B. *Multiple leakage barrier or subatmospheric containments.* The primary reactor containment barrier of a multiple barrier or subatmospheric containment shall be subjected to Type A tests to verify that its leakage rate meets the requirements of this appendix. Other structures of multiple barrier or subatmospheric containments (e.g., secondary containments for boiling water reactors and shield buildings for pressurized water reactors that enclose the entire primary reactor containment or portions thereof) shall be subject to individual tests in accordance with the procedures specified in the technical specifications, or associated bases.

V. Inspection and Reporting of Tests

A. *Containment inspection.* A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness. If there is evidence of structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, nondestructive examinations, and tests as specified in the applicable code specified in § 50.55a at the commencement of repair work. Such structural deterioration and corrective actions taken shall be reported as part of the test report, submitted in accordance with V.B.

B. *Report of test results.* 1. The preoperational and periodic tests shall be the subject of a summary technical report submitted to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 approximately 3 months after the conduct of each test. The report shall be titled "Reactor Containment Building Integrated Leak Rate Test."

2. The report on the preoperational test shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the preoperational test, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria.

3. For each periodic test, leakage test results from Type A, B, and C tests shall be reported. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test. Leakage test results from Type A, B, and C tests that

failed to meet the acceptance criteria of III.A.5(b), III.B.3, and III.C.3, respectively, shall be reported in a separate accompanying summary report that includes an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

[38 FR 4386, Feb. 14, 1973; 38 FR 5997, Mar. 6, 1973, as amended at 41 FR 16447, Apr. 19, 1976; 45 FR 62789, Sept. 22, 1980]

APPENDIX K—ECCS EVALUATION MODELS

I. Required and Acceptable Features of Evaluation Models.

II. Required Documentation.

I. REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODELS

A. Sources of heat during the LOCA.

For the heat sources listed in paragraphs 1 to 4 below it shall be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for such uncertainties as instrumentation error), with the maximum peaking factor allowed by the technical specifications. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime shall be studied and the one selected should be that which results in the most severe calculated consequences, for the spectrum of postulated breaks and single failures analyzed.

1. *The Initial Stored Energy in the Fuel.* The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

2. *Fission Heat.* Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.

3. *Decay of Actinides.* The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.

4. *Fission Product Decay.* The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards—"Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors". Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971). The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.

5. *Metal-Water Reaction Rate.* The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L. C., "Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, page 7, May 1962). The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall also be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less than 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

6. *Reactor Internals Heat Transfer.* Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

7. *Pressurized Water Reactor Primary-to-Secondary Heat Transfer.* Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors.)

B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.

The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

C. Blowdown Phenomena

1. *Break Characteristics and Flow.* a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

b. *Discharge Model.* For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," *Journal of Heat Transfer, Trans American Society of Mechanical Engineers,* 87, No. 1, February, 1965). The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperature calculated by this variation has been achieved.

c. *End of Blowdown.* (Applies Only to Pressurized Water Reactors.) For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the "end of bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining "end of bypass" include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

d. *Noding Near the Break and the ECCS Injection Points.* The noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

2. *Frictional Pressure Drops.* The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation (Baroczy, C. J., "A Systematic Correlation for Two-Phase Pressure Drop," *Chem. Enging. Prog. Symp. Series,* No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J.R.S., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," *Int. J. of Heat & Mass Transfer,* 7, 709–724, 1964) for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R. C. Nelson, D.B., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," *Transactions of ASME,* 695–702, 1948) for pressures lower than 250 psia is acceptable as a basis for calculating realistic two-phase friction multipliers.

3. *Momentum Equation.* The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational accelera-

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EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY UPPER-SHELF ENERGY LESS THAN 50 FT-LB.

CONTENTS

A. INTRODUCTION

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that the reactor vessel beltline materials ". . .must have Charpy upper- shelf energy of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." Charpy upper-shelf energy is defined in ASTM E 185-79 (Ref. 1) and -82 (Ref. 2), which are incorporated by reference in Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50. This guide describes general procedures acceptable to the NRC staff for demonstrating equivalence to the margins of safety in Appendix G of the ASME Code (Ref. 3). Several examples using these procedures are presented in Appendix A to this guide and in more detail in NUREG/CR-6023 (Ref. 4).

This regulatory guide contains information collections that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This regulatory guide has been submitted to the Office of Management and Budget for review and approval of the information collections. These information collections and record keeping are needed for demonstrating compliance with Appendix G to 10 CFR Part 50 for the remaining duration of the plant's license if Charpy upper-shelf energy of the materials in the beltline region may drop, or may have dropped, below the 50 ft-lb regulatory limit.

The public reporting burden for this collection of information is estimated to average 960 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for further reducing the reporting burden, to the Information and Records Management Branch (T6F33), U.S. Nuclear Regulatory Commission, Washington DC 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

B. DISCUSSION

The problem of evaluating materials that do not satisfy the 50 ft-lb upper-shelf energy requirement was recognized by the NRC staff several years ago and was designated Unresolved Safety Issue A-11, "Reactor Vessel Materials Toughness." In 1982, the staff completed resolution of USI A-11 by issuing NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue" (Ref. 5), which provided methods for evaluating the fracture behavior of these materials. Further, Generic Letter 82-26 (Ref. 6) was issued to advise licensees of the USI resolution. No new requirements were implemented as part of the USI resolution. However, neither NUREG-0744 nor Generic Letter 82-26 contained criteria for demonstrating equivalence of margins with Appendix G of the ASME Code. Rather, the NRC staff asked Section XI of the ASME Boiler Pressure Vessel Code Committee to develop and suggest to the staff appropriate criteria.

In February 1991, the Chairman of the ASME Section XI Subgroup on Evaluation and Standards provided to the NRC staff criteria that had been developed by members of the Working Group on Flaw Evaluation (WGFE) and the Working Group on Operating Plant Criteria (WGOPC) (Ref. 7). Although these criteria did not represent ASME Code criteria, they did represent the best opinion of knowledgeable persons familiar with the problem and with the ASME Code.

Upon review, the NRC staff found these criteria to be acceptable for demonstrating margins of safety equivalent to those in Appendix G of the ASME Code (Ref. 3). However, specific methods for evaluating the criteria still were being developed by the cognizant ASME Code committees. Further, those efforts were not expected to provide specific guidance on determining event sequences and transients to be considered, nor were they expected to provide specific guidance on appropriate material properties.

This guide has been developed to provide comprehensive guidance acceptable to the NRC staff for evaluating reactor pressure vessels when the Charpy upper-shelf energy falls below the 50 ft-lb limit of Appendix G to 10 CFR Part 50. The analysis methods in the Regulatory Position are based on methods developed for the ASME Code, Section XI, Appendix K (Ref. 8). The staff has reviewed the analysis methods in Appendix K and finds that they are technically acceptable but are not complete, because Appendix K does not provide information on the selection of transients and gives very little detail on the selection of material properties. In this regulatory guide, specific guidance is provided on selecting transients for consideration and on appropriate material properties to be used in the analyses.

Ductile tearing is the dominant fracture process in the upper-shelf region of the Charpy impact energy versus temperature curve for RPV materials. The conditions governing cleavage mode-conversion of the ductile tearing process in materials with low Charpy upper-shelf energy are still not well understood and are not considered in this regulatory guide.

The material property needed to characterize ductile tearing in the analysis methods in this regulatory guide is the material's J-integral fracture resistance, the J-R curve. This curve is a function of the material, the irradiation condition, the loading rate, and the material temperature. The curve is determined by testing the specific material, under the conditions of interest, in accordance with the American Society for

Testing and Materials Standard Test Method E 1152-87, "Standard Test Method for Determining J-R Curves" (Ref. 9).

Unfortunately, the specific material of interest (i.e., the material from the beltline region of the reactor vessel under operation) is seldom available for testing. Thus, testing programs have used generic materials that are expected to represent the range of actual materials used in fabricating reactor pressure vessels in the United States. Statistical analyses of these generic data have been performed and reported in NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data" (Ref. 10). These analyses provide a method for determining the material's J-integral fracture resistance that the NRC staff finds acceptable for use in the methods described in this guide. Other methods for determining the material property may be used on an individual-case basis if justified.

NOMENCLATURE

The following terms are used in this regulatory guide and its equations.

a	The flaw depth, which includes ductile flaw growth (in inches).
a_e	The effective flaw depth, which includes ductile flaw growth and a plastic-zone correction (in inches).
a_e^*	The effective stable flaw depth, which includes ductile flaw growth and a plastic-zone correction (in inches).
a_e^{**}	The effective stable flaw depth at tensile instability of the remaining ligament, which includes ductile flaw growth and a plastic-zone correction (in inches).
a_o	The postulated initial flaw depth (in inches).
2c	The total flaw length, which includes ductile flaw growth (in inches).
B_n	Net-section thickness of the ASTM E 1152-87 (Ref. 9) test specimen used in determining material tearing resistance, J-R curve, behavior (in inches).
C1, C2 C3, C4	Coefficients used in the equation for the material tearing resistance, J-R curve.
CR	The cooldown rate (°F/hour).
CVN	Charpy v-notch upper-shelf energy (ft-lb.).
E	Young's modulus of elasticity (ksi).
E'	$E/(1-\nu^2)$ (ksi).
F_1, F_2, F_3	Geometry factors used to calculate the stress intensity factors (dimensionless).
$J_{applied}$	The J-integral from the applied loads (in.-lb/in.^{2}).
$J_{material}$	The material's J-integral fracture resistance (in.-lb/in.^{2}), J-R curve.
$J_{0.1}$	The material's J-integral fracture resistance at a ductile flaw growth of 0.10 in. (in.-lb/in.^{2}).
K_{IcL}	The mode I stress intensity factor caused by the radial thermal gradient through the cladding applied to the vessel inner surface, calculated with no plastic zone correction (ksi $\sqrt{\text{in.}}$).
K_{Ip}	The mode I stress intensity factor caused by the internal pressure, calculated with no plastic-zone correction (ksi $\sqrt{\text{in.}}$); K_{Ip}^{Axial} and $K_{Ip}^{Circum.}$ are the axial and circumferential values, respectively.
K'_{Ip}	K_{Ip} calculated with a plastic-zone correction (ksi $\sqrt{\text{in.}}$).
K_{It}	The mode I stress intensity factor caused by the radial thermal gradient through the vessel wall, calculated with no plastic-zone correction (ksi $\sqrt{\text{in.}}$).
K'_{It}	K_{It} calculated with a plastic-zone correction (ksi $\sqrt{\text{in.}}$).
p	Internal pressure (ksi).
p_a	The maximum accumulation pressure as defined in the plant-specific Overpressure Protection Report, but not exceeding 1.1 times the design pressure (ksi).
R_i	The inner radius of the vessel (in inches).
SF	The safety factor (dimensionless).
t	The wall thickness of the vessel's base metal (in inches).
t'	The sum of the vessel wall thickness, t, and the cladding thickness, t_{cL} (in inches).
t_{cL}	The thickness of the stainless steel cladding applied to the vessel inner surface (in inches).
T	Metal temperature, at crack-tip, used in the analysis (°F).

MF The margin factor = 2 standard deviations on test data (dimensionless).

σ_f A reference material's flow stress, specified as 85 ksi in ASME Section XI, Appendix K (Ref. 8), on Charpy upper-shelf energy.

σ_y The material's yield stress (ksi).

ν Poisson's ratio (dimensionless), specified as 0.3.

C. REGULATORY POSITION

1. ACCEPTANCE CRITERIA

The following criteria are acceptable to the NRC staff for demonstrating that the margins of safety against ductile fracture are equivalent to those in Appendix G to Section III of the ASME Code. Licensees may follow this regulatory guide to determine the equivalent safety margins, or they may use any other methods, procedures, or selection of materials data and transients to demonstrate compliance with Appendix G to 10 CFR Part 50. If licensees choose to follow this regulatory guide, they must use the acceptance criteria, analysis methods, material properties, and selection of transients as described in this regulatory guide. The acceptance criteria are to be satisfied for each category of transients, namely, Service Load Levels A and B (normal and upset), Level C (emergency), and Level D (faulted) conditions. These service load levels are described in Standard Review Plan 3.9.3 (Ref. 11). Because of differences in acceptable outcome during the various sevice load levels, different criteria have been developed for Levels A and B, C, and D.

1.1 Level A and B Conditions

When the upper-shelf Charpy energy of the base metal is less than 50 ft-lb, postulate both axial and circumferential interior flaws and use the toughness properties for the corresponding orientation. For a weld with Charpy upper-shelf energy less than 50 ft-lb, postulate an interior surface flaw oriented along the weld of concern and orient the flaw plane in the radial direction. Postulate a semi-elliptical surface flaw with an a/t = 0.25 and with an aspect ratio of 6-to-1 surface length to flaw depth. A smaller flaw size may be used on an individual-case basis if justified. Two criteria must be satisfied as described below. The maximum accumulation pressure, discussed below, is the maximum pressure defined in the Over Pressure Protection Report that satisfies the requirement of Section III, NB-7311(b), of the ASME Code (Ref. 12).

1.1.1 The crack driving force must be shown to be less than the material toughness as given by Equation 1:

$$J_{applied} < J_{0.1} \qquad (1)$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw under pressure and thermal loading where the assumed pressure is 1.15 times the maximum accumulation pressure, with thermal loading using the plant-specific heatup and cooldown conditions. The parameter $J_{0.1}$ is the J-integral characteristic of the material's resistance to ductile tearing ($J_{material}$), as denoted by a J-R curve test, at a crack extension of 0.1 inch.

1.1.2 The flaw must be stable under ductile crack growth as given by Equation 2:

$$\frac{\partial J_{applied}}{\partial a} < \frac{\partial J_{material}}{\partial a} \qquad (2)$$

(with load held constant)

at

$$J_{applied} = J_{material}$$

where $J_{applied}$ is calculated for the postulated flaw under pressure and thermal loading for all service level A and B conditions where the assumed pressure is 1.25 times the maximum accumulation pressure, with thermal loading, as defined above. The material's J-integral fracture resistance should represent a conservative estimate of the data for the vessel material under evaluation (i.e., mean - 2 standard deviations). Methods for determining the J-integral fracture resistance, J-R curve, are discussed in Regulatory Position 3 of this guide. Methods for determining the appropriate service level conditions are discussed in Regulatory Position 4 of this guide.

1.2 Level C Condition

When the Charpy upper-shelf energy of the base metal is less than 50 ft-lb, postulate both axial and circumferential interior flaws and use the toughness properties for the corresponding orientation. When the Charpy upper-shelf energy of any weld material is less than 50 ft-lb, postulate an interior surface flaw with its major axis oriented along the weld of concern and the flaw plane oriented in the radial direction. Consider postulated surface flaws with depths up to one-tenth the base metal wall thickness, plus the clad thickness, but with the total depth not to exceed 1.0 inch (2.54 cm) and with an aspect ratio of 6-to-1 surface length to flaw depth. A smaller maximum flaw depth may be used on an individual-case basis if justified. For these evaluations, two criteria must be satisfied.

1.2.1 The crack driving force must be shown to be less than the material toughness as given by Equation 3:

$$J_{applied} < J_{0.1} \qquad (3)$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw in the beltline region of the reactor vessel under the governing Service Level C condition, with a safety factor of 1.0 on the applied loading. $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as denoted by a J-R curve test, at a crack extension of 0.1 inch.

1.2.2 The flaw must also be stable under ductile crack growth as given by Equation 4:

$$\frac{\partial J_{applied}}{\partial a} < \frac{\partial J_{material}}{\partial a} \quad (4)$$

(with load held constant)

at

$$J_{applied} = J_{material}$$

where $J_{applied}$ is calculated for the postulated flaw under the governing Service Level C condition, with a safety factor of 1.0 on the applied loading. The material's J-integral fracture resistance should represent a conservative estimate of the data for the vessel material under evaluation (i.e., mean - 2 standard deviations). The J-integral resistance versus crack growth, J-R curve, is defined in Regulatory Position 3 of this guide. Determination of the appropriate service level conditions is discussed in Regulatory Position 4 of this guide.

1.3 Level D Condition

When the Charpy upper-shelf energy of the base metal is less than 50 ft-lb, postulate both axial and circumferential interior flaws and use the toughness properties for the corresponding orientation. When the Charpy upper-shelf energy of any weld material is less than 50 ft-lb, postulate an interior semi-elliptic surface flaw with the major axis oriented along the weld of concern and the flaw plane oriented in the radial direction. Consider postulated surface flaws with depths up to one-tenth the base metal wall thickness, plus the clad thickness, but with total depth not to exceed 1.0 inch (2.54 cm) and with an aspect ratio of 6-to-1 surface length to flaw depth. A smaller maximum flaw depth may be used on an individual case basis if justified.

For these evaluations, the postulated flaw must be stable under ductile crack growth as given by Equation 5:

$$\frac{\partial J_{applied}}{\partial a} < \frac{\partial J_{material}}{\partial a} \quad (5)$$

(with load held constant)

at

$$J_{applied} = J_{material}$$

where $J_{applied}$ is calculated for the postulated flaw under the governing Service Level D condition, with a safety factor of 1.0 on the applied loading. Additionally, the flaw depth, including stable tearing, should not be greater than 75% of the vessel wall thickness, and the remaining ligament should be safe from tensile instability. The material's J-integral fracture resistance should reflect a best estimate, i.e., the mean value, of the data representative of the vessel material under evaluation.

The J-integral resistance versus crack growth, J-R curve, is discussed in Regulatory Position 3 of this guide. Methods for determining the appropriate service level conditions are discussed in Regulatory Position 4 of this guide.

2. ANALYSIS METHODS

The analysis methods described in this guide are acceptable to the NRC staff for evaluating the criteria described above. Other methods may be used if justified on a case-by-case basis.

2.1 Level A and B Conditions

The acceptance criteria discussed in Regulatory Position 1.1 for Level A and B conditions involve a comparison of the applied J-integral to the material's J-integral fracture resistance at a ductile flaw extension of 0.1 inch and a determination that this flaw would be stable under the applied loading. Procedures are detailed below for (1) calculating the applied J-integral for Service Levels A and B flaws and loading conditions and (2) determining that the slope of the material's J-integral resistance curve is greater than the slope of the applied J-integral versus crack depth curve at the equilibrium point on the J-R curve where the two curves intersect, as illustrated in Figure 1.

2.1.1 Calculation of the Applied J-Integral

The calculation of the applied J-integral consists of two steps: Step 1 is to calculate the effective flaw depth, which includes a plastic-zone correction, and Step 2 is to calculate the J-integral for small-scale yielding based on this effective flaw depth.

Step 1

For an axial flaw with depth 'a' equal to (0.25t + 0.1 in.), calculate the stress intensity factor from internal pressure, p_a, with a safety factor, SF, on pressure equal to 1.15, using Equation 6:

$$K_{Ip}^{Axial} = (SF)\, p_a\, [1 + (R_i/t)]\, (\pi a)^{0.5}\, F_1 \quad (6)$$

$$F_1 = 0.982 + 1.006(a/t)^2$$

This equation for K_{Ip}^{Axial} is applicable to $0.05 \leq a/t \leq 0.50$, and it includes the effect of pressure acting on the flaw faces.

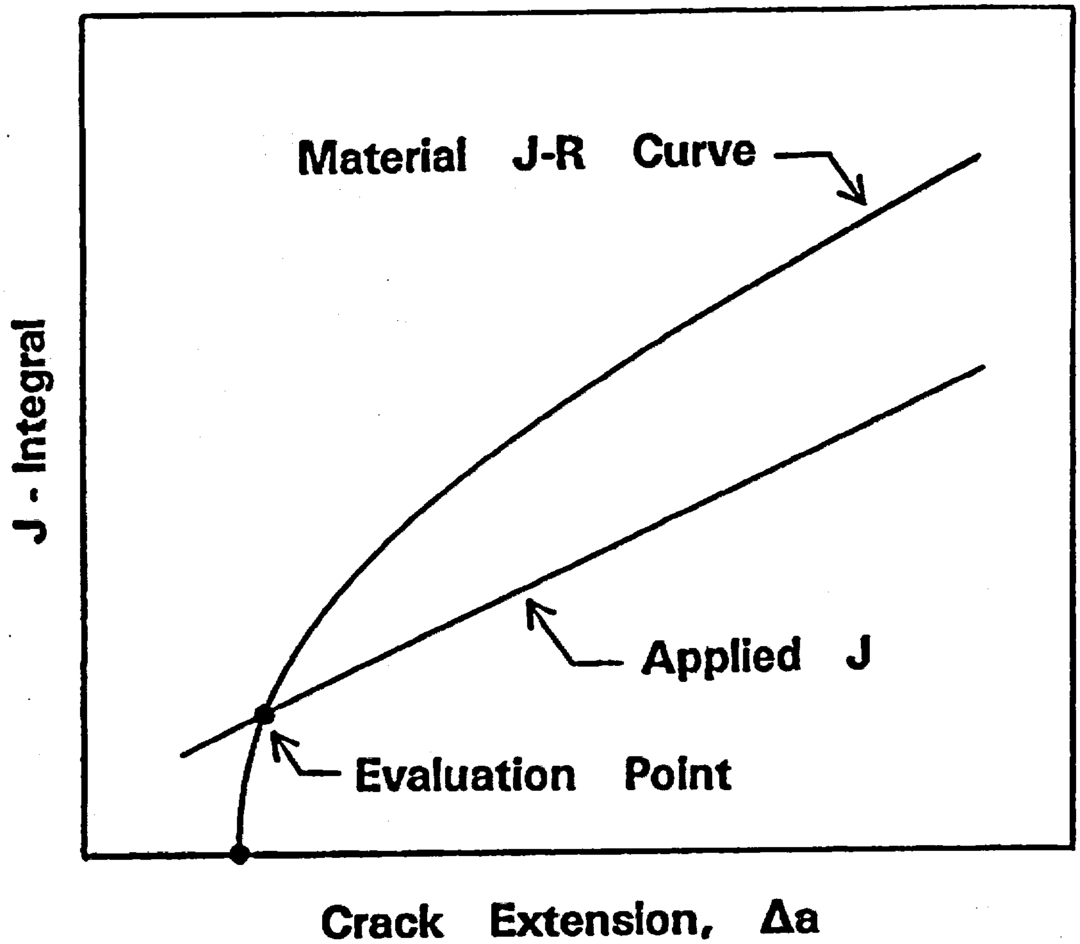

Figure 1. Comparison of the Slope of the Applied J-Integral and J-R Curve.

For a circumferential flaw with depth 'a' equal to (0.25t + 0.1 in.), calculate the stress intensity factor from internal pressure, p_a, with a safety factor, SF, on pressure equal to 1.15, using Equation 7:

$$K_{Ip}^{Circum.} = (SF)p_a[1+(R_i/(2t))](\pi a)^{0.5}F_2 \tag{7}$$

$$F_2 = 0.885 + 0.233(a/t) + 0.345(a/t)^2$$

This equation for $K_{Ip}^{Circum.}$ is applicable to $0.05 \le a/t \le 0.50$, and it includes the effect of pressure acting on the flaw faces.

For an axial or circumferential flaw with depth 'a' equal to (0.25t + 0.1 in.), the "steady-state" (time independent) stress intensity factor from radial thermal gradients is obtained by using Equation 8:

$$K_{It} = ((CR)/1000)t^{2.5} F_3 \tag{8}$$

$$F_3 = 0.69 + 3.127(a/t) - 7.435(a/t)^2 + 3.532(a/t)^3$$

This equation for K_{It} is valid for $0.2 \le a/t \le 0.50$, and $0 \le CR \le 100$°F/hr. This equation does not include the contribution to K_{It} from the cladding thickness, t_{cL}. If the steady-state values of thermally induced K_{It} are used, the material J-R curve should correspond to the temperature at the beginning of the transient, when a uniformly high temperature is present across the vessel wall thickness, leading to the lowest J-R curve. The above K_{It} expression can be replaced with an improved accuracy solution if an appropriate justification is provided.

Calculate the effective flaw depth for small-scale yielding, a_e, using Equation 9:

$$a_e = a + \left(\frac{1}{6\pi}\right)\left[\frac{(K_{Ip}+K_{It})}{\sigma_y}\right]^2 \tag{9}$$

Step 2

For an axial flaw, calculate the stress intensity factor from internal pressure for small-scale yielding, K'_{Ip}, by substituting a_e in place of 'a' in Equation 6, including the equation for F_1. For a circumferential flaw, calculate K'_{Ip} by substituting a_e in place of 'a' in Equation 7, including the equation for F_2. For an axial or circumferential flaw, calculate the stress intensity factor from the radial thermal gradients for small-scale yielding K'_{It}, by substituting a_e in place of 'a' in Equation 8, including the equation for F_3.

The J-integral from the applied loads for small-scale yielding is given by Equation 10:

$$J_{applied} = 1000(K'_{Ip}+K'_{It})^2 / E' \tag{10}$$

Alternatively, in place of the steady-state Equation 8, a thermal transient stress analysis may be performed for the limiting cooldown rate, including the contributions of cladding to thermal stress and the thermal stress intensity factor. For this alternative analysis method (also described in Reference 4), the main features for computing K_{It} and K_{ItL}, which are applied in examples in Appendix A, are given in Appendix B.[1,2] The limiting condition should be determined for the transient time at which the material's J-R curve will be greater than or equal to the $J_{applied}$ for evaluating Equations 1 and 2. The main steps are:

a. Determine the temperature gradient across the vessel wall thickness, in 10 to 20 time steps over the full duration of the transient; and compute the corresponding thermal stress history, taking into account the cladding thickness, t_{cL}.
b. For each time step, compute K_{It} and K_{ItL} values as a function of the crack depth in the range $0.05 \le a/t' \le 0.5$.
c. For Equation 1, calculate the pressure-induced K_{Ip} and the $J_{applied}$, using Equations 9 and 10, at a crack-tip depth of (0.25t' + 0.1 in.) for each time step.
d. Use Step a to find crack-tip temperature history at each time step. See Figure A-1 in Appendix A for an example.
e. For a given material condition, determine the J-R values at the crack extension of 0.1 inch by using the crack-tip temperature history from Step d. See Figure A-2 in Appendix A for an example.
f. Compare the material's J-R values as a function of time in Step e with the $J_{applied}$ values in Step c. See Figure A-2 in Appendix A for an example. The time at which the J-R value is just equal to the $J_{applied}$ determines the critical condition for evaluating Equation 1.
g. At the time determined in Step f, evaluate Equation 2 to verify the stability of the predicted flaw growth.

2.1.2 Evaluation of Flaw Stability

Flaw stability is evaluated by a direct application of the flaw stability criterion given by Equation 2. The applied J-integral is calculated for a series of flaw depths corresponding to increasing amounts of ductile flaw growth. The applied pressure, p, is set equal to the maximum accumulated pressure for Service Level A and B conditions, p_a, with a safety factor, SF, equal to 1.25. The applied J-integral for Service Level A and B conditions may be calculated using Equations 6 through 10. Each pair of the applied J-integral and flaw depth is plotted on a crack driving force diagram to produce the

[1] The equations provided in Appendix B may be used if the transient temperature history can be approximated adequately by either an exponential or a polynomial equation. If it cannot be approximated adequately, a more rigorous approach should be used.

[2] The computer code given in Appendix B is for general illustration. Licensees assume responsibility for the correctness of the computer codes they use.

applied J-integral curve as illustrated in Figure 1. The material's J-R curve also is plotted on the crack driving force diagram. Flaw stability at a given applied load is demonstrated if the slope of the applied J-integral curve is less than the slope of the material's J-R curve at the equilibrium point on the J-R curve where the two curves intersect.

2.2 Level C Condition

The acceptance criteria discussed in Regulatory Position 1 for Service Level C conditions are similar to those for Service Levels A and B, with the exceptions of the crack size to be considered and the safety factor applied to the pressure loading. For Service Level C conditions, flaw sizes up to one-tenth the base metal wall thickness, plus the clad thickness t_{cL}, but with a total depth not to exceed 1.0 inch (2.54 cm), are to be considered. A safety factor of 1.0 is used for both pressure and thermal loading. As with the Service Level A and B criteria, for Service Level C it must be demonstrated that the applied J is less than the material's fracture resistance at a crack extension of 0.1 inch, and that the flaw must be stable under the applied loading.

Procedures are described below for (1) determining the applied J-integral for Service Level C flaw and loading conditions and (2) determining that the slope of the material's J-integral fracture resistance, J-R curve, is greater than the slope of the applied J-integral versus crack depth curve.

2.2.1 Calculation of the Applied J-Integral

The calculation of the applied J-integral consists of two steps: Step 1 is to calculate the effective flaw depth, which includes a plastic-zone correction, and Step 2 is to calculate the J-integral for small-scale yielding based on this effective flaw depth.

Step 1

Postulate a series of flaws with depths ranging up to cladding thickness plus 0.1 times the base metal wall thickness, but not exceeding 1.0 inch (2.54 cm). The number of flaws and the specific flaw sizes to be postulated should be sufficient to determine the peak value of the applied J-integral over this size range. For each of these postulated flaws, the analysis flaw size 'a' should be the sum of the postulated flaw size plus 0.1-inch ductile crack extension. For axial flaws, at each analysis flaw size, calculate the stress intensity factor arising from internal pressure, p_a, with a safety factor, SF, on internal pressure equal to 1.0, using Equation 11:

$$K_{Ip}^{Axial} = (SF)p_a\ [1 + (R_i/t')]\ (\pi a)^{0.5} F_1 \qquad (11)$$

$$F_1 = 0.982 + 1.006(a/t')^2;\ with\ 0.05 \le a/t' \le 0.5$$

For circumferential flaws, at each analysis flaw size calculate the stress intensity factor arising from internal pressure, p_a, with a safety factor, SF, on pressure equal to 1.0, using Equation 12:

$$K_{Ip}^{Circum.} = (SF)p_a[1 + R_i/(2t')](\pi a)^{0.5} F_2 \qquad (12)$$

$$F_2 = 0.885 + 0.233(a/t') + 0.345(a/t')^2$$

These equations for $K_{Ip}^{Circum.}$ are valid for $0.05 \le a/t' \le 0.5$, and include the effect of pressure acting on the flaw faces.

If it can be demonstrated that the actual cooldown rate could be bounded by a "constant" cooldown rate, for each crack depth the stress intensity factor arising from radial thermal gradient, including cladding effects (see Example 4 in Appendix A) is given by Equation 13:

$$\begin{aligned} K_{It} = [&-0.012771 + 0.849528(\frac{CR}{1000}) - 0.611382(\frac{CR}{1000})^2 \\ &+ (0.565188 + 0.0467582(\frac{CR}{1000}))(\frac{a}{t'}) - 1.85371(\frac{a}{t'})^2 \\ &+ 1.62878(\frac{a}{t'})^3](t')^{2.5} \end{aligned} \qquad (13)$$

This equation is applicable to $0.05 \le a/t' \le 0.5$, and $100 \le CR \le 600°F/hour$. The CR values less than 100°F/hour are covered under Service Levels A and B (see Equation 8). The cladding thickness is $t_{cL} = 5/16$ in., $R_i = 86.875$ in., base metal thickness t = 8.625 in., and R_i/t' ratio = 9.72. Details of the analysis results are given in Appendix A. Equation 13 is based on the current state of knowledge on K solutions for 6:1 aspect-ratio flaws subjected to non-uniform stress gradients in the crack-depth direction. The above K_{It} expression can be replaced with an improved accuracy solution if an appropriate justification is provided.

Calculate the effective flaw depth for small-scale yielding, a_e, using Equation 14:

$$a_e = a + (\frac{1}{6\pi})\ [\frac{(K_{Ip} + K_{It})}{\sigma_y}]^2 \qquad (14)$$

Step 2

For each flaw size considered, calculate the stress intensity factor arising from internal pressure for small-scale yielding, K'_{Ip}, by substituting a_e in place of 'a' in Equation 11 for the axial flaws and in Equation 12 for the circumferential flaws. Similarly, calculate the stress intensity factor arising from radial thermal gradients for small-scale yielding, K'_{It}, by substituting a_e in place of 'a' in Equation 13. The J-integral arising from the applied loads for small-scale yielding is given by Equation 15:

$$J_{applied} = 1000(K'_{Ip} + K'_{It})^2\ /E' \qquad (15)$$

In an actual transient the cooldown rate initially may vary significantly with time. Therefore, transient-specific peak thermal stress-induced K_{It} and K_{IcL} computations may be necessary. If so, in place of Equation 13, a thermal transient stress analysis may be performed for the specific transient, including the contributions of cladding to thermal stress and the stress intensity factor. For this alternative analysis method the main features for computing K_{It} and K_{IcL}, which are applied on examples in Appendix A, are given in Appendix B.[1,2] The limiting condition should be determined for the transient time at which the material's resistance (J-R curve) will be greater than or equal to the $J_{applied}$ for evaluating Equations 1 and 2. The main steps are:

a. Determine the temperature gradient across the vessel wall thickness, in 10 to 20 time steps over the full duration of the transient, and compute the corresponding thermal stress history, taking into account the cladding thickness, t_{cL}.
b. For each time step, compute K_{It} and K_{IcL} values as a function of the crack depth in the range $0.05 \le a/t' \le 0.5$.
c. For Equation 1, calculate the pressure-induced K_{Ip} and the $J_{applied}$, using Equations 14 and 15, at a crack-tip depth of $\{(0.1t + t_{cL} + 0.1 \text{ in.}) \le 1 \text{ in.}\}$ for each time step.
d. Use Step a to find crack-tip temperature history at each time step. See Figure A-1 in Appendix A for an example.
e. For a given material condition, determine the J-R values at the crack extension of 0.1 inch by using the crack-tip temperature history from Step d. See Figure A-2 in Appendix A for an example.
f. Compare the material's J-R values as a function of time in Step e with the $J_{applied}$ values in Step c. See Figure A-2 in Appendix A for an example. The time at which the J-R value is just equal to the $J_{applied}$ determines the critical condition for evaluating Equation 1.
g. At the time determined in Step f, evaluate Equation 2 to verify the stability of predicted flaw growth.

2.2.2 Evaluation of Flaw Stability

Flaw stability is evaluated by a direct application of the flaw stability criterion given by Equation 4. The applied J-integral is calculated for a series of flaw depths corresponding to increasing amounts of ductile flaw growth. The applied pressure, p, is set equal to the peak pressure for the Service Level C transient under consideration with a safety factor, SF, equal to 1.0. The applied J-integral for Service Level C conditions may be calculated using Equations 11 through 15. Each pair of the applied J-integral and flaw depth is plotted on a crack driving force diagram to produce the applied J-integral curve as illustrated in Figure 1. The material's J-R curve also is plotted on the crack driving force diagram and intersects the abscissa at the initial flaw depth, a_o. Flaw stability at a given applied load is demonstrated if the slope of the applied J-integral curve is less than the slope of the material's J-R curve at the equilibrium point on the J-R curve where the two curves intersect.

2.3 Level D Condition

The acceptance criteria discussed in Regulatory Position 1 for Level D Service Conditions involve only the stability of the postulated flaws. Additionally, the stable flaw depth must not exceed 75% of the vessel wall thickness, and the remaining ligament must be safe from the tensile instability.

Stability of ductile crack extension is demonstrated for Service Level D in the same manner used for Service Level C. However, the material properties should represent only the best estimate (i.e., mean value) of the J-R curve for the vessel material under evaluation.

Tensile stability of the remaining ligament is conservatively demonstrated if Equation 16 is satisfied.

$$\sigma_f > 2p(R_i + a_e^{**}) / [\sqrt{3}(t - a_e^{**})] \qquad (16)$$

Where, from Reference 13, for a semi-elliptical flaw,

$$a_e^{**} = [a_e^*(1 - \{1 + 2c^2/t^2\}^{-0.5})] \,/\, [1 - (a_e^*/t)\{1 + 2c^2/t^2\}^{-0.5}]$$

3. MATERIAL PROPERTIES

The statistical analyses reported in Reference 10 addressed a broad range of materials and conditions. For the purposes of this guide, the NRC staff has concluded that only the ASTM E 1152-87 (Ref. 9) definition of the J-integral fracture resistance curve should be used. This determination requires that a test specimen's net thickness, B_n, be specified. Smaller specimens typically produce more conservative (lower) J-R curves than larger specimens. However, larger specimens are needed to provide large amounts of crack growth needed in evaluating certain stability criteria described in Regulatory Position 2 of this regulatory guide. The NRC staff recommends the test specimen's net-section thickness, B_n, to be 1.0 inches (2.54 cm) for determining the J-integral resistance curve using the methods specified in Regulatory Position 3. This is a reasonable compromise and slightly simplifies the equations for the material J-R curve. The neutron fluence attenuation at any depth in the vessel wall (such as near the crack tip) should be determined using Regulatory Guide 1.99 (Ref. 14).

This guide provides methods for determining the J-integral fracture resistance of three classes of materials: welds manufactured with Linde 80 welding flux, generic welds used in fabricating reactor pressure vessels, and plate materials (low and high toughness). The J-R curves for plant-specific materials may be used if justified on a case-by-case basis. Otherwise, the material's J-integral fracture resistance may be determined from Equation 17, developed in Reference 10:

$$J_R = (MF)\ \{C1(\Delta a)^{C2} \exp[C3\ (\Delta a)^{C4}]\} \qquad (17)$$

The coefficients in Equation 17 for each material type are discussed below. As noted earlier, the net-section thickness, B_n, of ASTM E 1152-87 (Ref. 9) compact-tension (CT) specimens to be considered is specified as 1 inch. In addition to the Charpy (CVN) models discussed in this guide, Reference 10 contains two other models, namely the Copper-Fluence (Cu-ϕt) models and the pre-irradiation Charpy (CVN_p) models, which may be used to determine the material's J-R curves.

3.1 Welds Made Using Linde 80 Flux

For analyses addressing Service Levels A, B, and C, a conservative representation of the J-R curve is obtained by setting the margin factor, MF = 0.648. For analyses addressing Service Level D, set MF = 1.0.

$$C1 = \exp[-3.67+1.45\ \ln(CVN) - 0.00308T] \tag{18}$$

$$C2 = 0.077 + 0.116\ \ln C1 \tag{19}$$

$$C3 = -0.0812 - 0.0092\ \ln C1 \tag{20}$$

$$C4 = -0.5 \tag{21}$$

3.2 Generic Reactor Pressure Vessel Welds

For analyses addressing Service Levels A, B, and C, a conservative representation of the J-R curve is obtained by setting the margin factor, MF = 0.629. For analyses addressing Service Level D, set MF = 1.0.

$$C1 = \exp[-4.12+1.49\ \ln(CVN) - 0.00249T \tag{22}$$

$$C2 = 0.077 + 0.116\ \ln C1 \tag{23}$$

$$C3 = -0.0812 - 0.0092\ \ln C1 \tag{24}$$

$$C4 = -0.5 \tag{25}$$

3.3 Reactor Pressure Vessel Base (Plate) Materials

The elastic-plastic fracture toughness of plate materials may be relatively high or quite low, depending on a variety of chemical, metallurgical, and thermo-mechanical processing variables. The statistical analyses reported in Reference 10 included only materials that exhibited a J-R curve with a significantly rising slope, i.e., the higher toughness materials. However, test results reported in NUREG/CR-5265, "Size Effects on J-R Curves for A-302B Plate" (Ref. 15), clearly show J-R curves with very little, if any, increase in slope. References 15, 16, and 17 provide some insight into the nature of the low toughness issue for the plate materials. While there are several variables that influence the fracture toughness, sulphur content seems to be a reasonable indicator of the plate toughness, with a "higher" sulphur content indicating "lower" fracture toughness (Ref. 17). A sulphur content of 0.018 wt-% is a good demarcation for high- and low-toughness values.

Because of the low-toughness plate issue, and because of the relatively sparse data base that could be used to estimate the fracture toughness for these materials, a fracture toughness model is only provided for high-toughness plate materials. If the sulphur content of the plate is less than 0.018 wt-%, the plate models described in Reference 10 may be used. However, if the sulphur content is greater than or equal to 0.018 wt-%, justification should be provided for use of the models in Reference 10. Factors that might justify use of these high-toughness models could include information about the year of manufacture of the plate and any special thermo-mechanical processing that would serve to improve the fracture toughness of the plate. If adequate justification cannot be provided, a low-toughness plate model should be developed and used.

The CVN value should be for the proper orientation of the plate material (see Figure 2). For example, for axial flaws the CVN value for the L-T (strong) orientation in the vessel wall should be used. Similarly, for circumferential flaws the CVN value for the T-L (weak) orientation should be used. In many cases, the CVN values for both orientations may not be known. If the CVN value for the T-L (weak) orientation is not available, the L-T (strong) orientation CVN value may be multiplied by a factor of 0.65 (Ref. 18) to obtain the CVN value for the T-L (weak) orientation. However, if the CVN value for the T-L (weak) orientation is known and the L-T (strong) orientation is to be estimated, the CVN value for the L-T (strong) orientation is assumed to be the same as that of the T-L (weak) orientation.

3.3.1 High-Toughness Model (S < 0.018 Wt-%)

For plate material with sulphur content greater than 0.018 wt-%, the use of this model should be justified as discussed above.

For analyses addressing Service Levels A, B, and C, a conservative representation of the J-R curve is obtained by setting the margin factor, MF = 0.749. For analyses addressing Service Level D, set MF = 1.0.

$$C1 = \exp[-2.44+1.13\ \ln(CVN) - 0.00277T] \tag{26}$$

$$C2 = 0.077 + 0.116\ \ln C1 \tag{27}$$

$$C3 = -0.0812 - 0.0092\ \ln C1 \tag{28}$$

$$C4 = -0.409 \tag{29}$$

DEFINITION OF ASME AND ASTM ORIENTATIONS

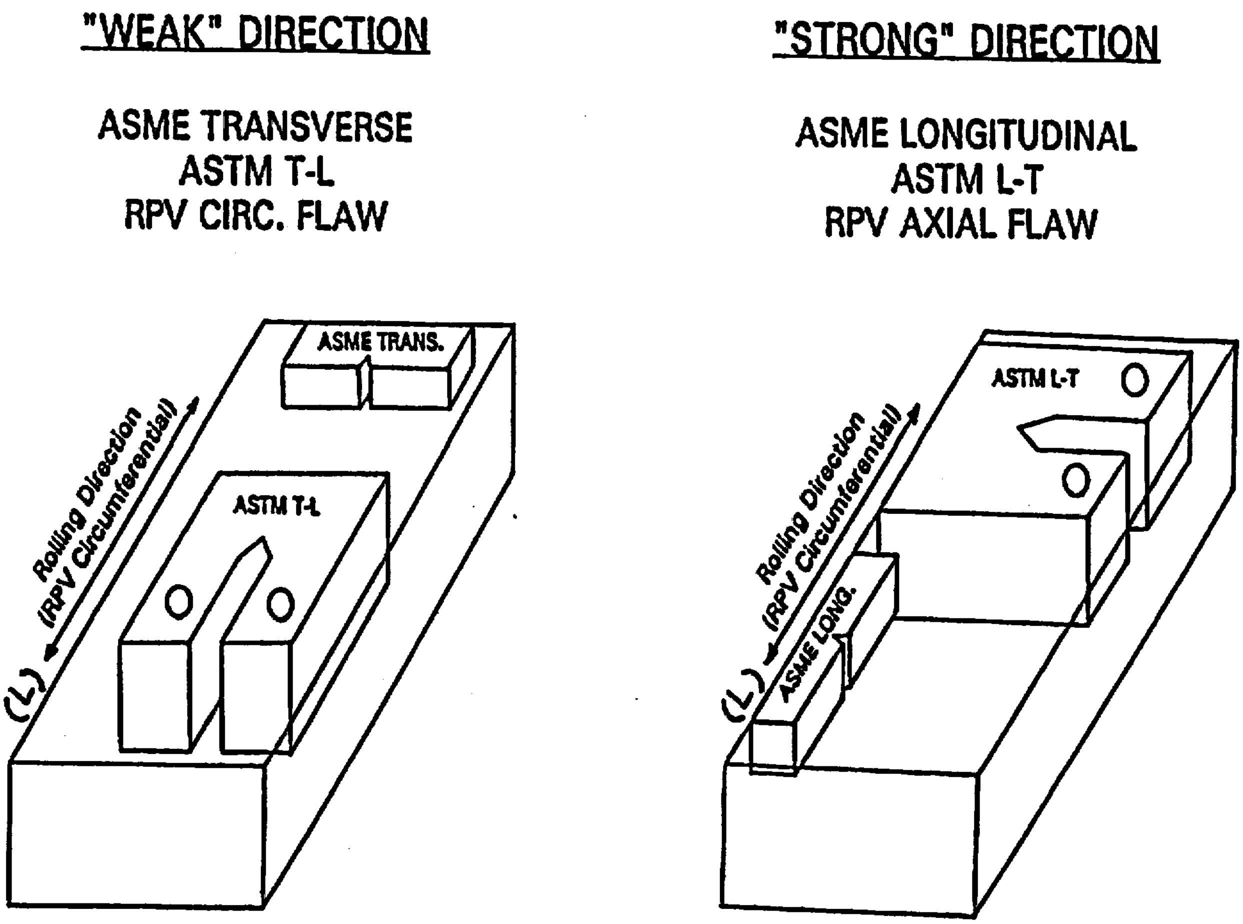

Figure 2. Definition of the ASME and ASTM Flaw Orientations in an RPV.

3.3.2 Low-Toughness Plate (S ≥ 0.018 Wt-%)

For analyses addressing materials with a sulphur content greater than 0.018 wt-%, the J-R curve data are scarce. Very limited J-R data for a 6-inch-thick specimen (ASTM 6T CT at 180°F temperature) from an A-302B plate in the T-L (weak) orientation, available in NUREG/CR-5265 (Ref. 15), may be used with adjustments for the specimen temperature and CVN value (Ref. 19), or a material-specific justification should be provided to support the use of other data. For analyses addressing Service Levels A, B, and C, a lower-bound representation (mean - 2 standard deviations) of the J-R curve should be used. For analyses addressing Service Level D, the mean value of the J-R curve should be used.

Additional J-R curve test data for the low-toughness A302B plate material are presently being generated. Regulatory guidance will be updated, if justified, based on the results obtained from the test data collected for J-R curve in low-toughness plate material.

4. TRANSIENT SELECTION

Selection of the limiting transients for Service Levels C and D is a key aspect of evaluating the integrity of reactor pressure vessels that contain materials with Charpy upper-shelf energy less than 50 ft-lb. Generally, Service Levels A and B are limiting. However, there may be plant-specific considerations that make Service Levels C or D controlling for ductile fracture.

To provide reasonable assurance that the limiting service loading conditions have been identified, either of two approaches may be used: a plant-specific transient evaluation or a generic bounding analysis. It should be noted that plants may be grouped and limiting transients for these groups may be determined. The plant-specific transient evaluation is the preferred approach. However, since some licensees may not have the specific transient information needed for this analysis, a conservative "bounding" analysis may be performed for each service level. Specific guidance for each of these approaches is provided below.

As described in the Discussion section of this guide, ductile tearing is the dominant fracture process in the upper-shelf region, and the possibility of mode-conversion to cleavage (brittle) fracture is not considered in this regulatory guide. The analyses using these bounding transients need only address the transient from its beginning to the time at which the metal at the tip of the flaw being analyzed reaches a temperature equivalent to the adjusted RT_{NDT} plus 50°F. In this regulatory guide, an adjusted RT_{NDT} plus 50°F (which typically represents the low-temperature overpressure protection system's enabling temperature) is taken as the lower temperature limit for upper-shelf behavior.

This regulatory guide states that licensees should consider a spectrum of transients, including ATWS (anticipated transient without scram). Although ATWS is not a design basis transient, for compliance with Appendix G to 10 CFR Part 50 it was considered in Reference 4 for evaluation of low upper-shelf energy materials. Based on the generic analyses in Reference 4 and additional staff calculations, ATWS in currently operating light-water-reactor (LWR) vessels in the United States is not found to be a dominant transient with respect to the low Charpy upper-shelf energy issue, and no further action is necessary with respect to ATWS. However, for designs other than the currently operating LWR vessels in the United States, ATWS could become a dominating transient, and as such needs to be considered as a Service Level C transient for further evaluation. A plant-specific justification should be provided for consideration of such designs at another service load level. For such designs, licensees should consider the assumptions used in the generic analyses of Reference 4 to be sure that they are bounding for their plant-specific applications. If these generic analyses are not bounding, plant-specific analyses should be performed.

4.1 Plant-Specific Transients

To provide reasonable assurance that the limiting service loading conditions have been identified on a plant-specific basis, the Service Level C and D design transients and events that are necessary to demonstrate compliance with Standard Review Plan 3.9.3 (Ref. 11) should be used.

When this transient list is not available or is incomplete, the most complete list of transients for these service levels that is available for similar plant designs should be used. Typically, the most complete list of transients would be for the later-vintage plants from a particular vendor. This list should be reviewed, and the limiting transients for the reactor vessel being analyzed should be defined. Once the transients are defined, system-level thermal-hydraulic analyses should be performed to determine the limiting pressure-temperature-time history for each transient being considered. This history provides the input to the analyses described in this guide.

4.2 Bounding Transients

When the plant-specific transients are not available or when developing or updating the pressure-temperature-time history would be an undue burden, a conservative "bounding" pressure-temperature-time history may be used. This history should anticipate a pressure equal to the shut-off head for the high-pressure injection system and a cooldown rate of 400°F per hour for Service Level C and 600°F per hour for Service Level D. These values are based on the NRC staff's experience in performing the bounding analyses (for examples, see Appendix A of this regulatory guide and Reference 4). Alternatives to these cooldown rates may be used if justified by the plant-specific safety-injection flows and temperatures.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide reflecting public comments will be used by the NRC staff in the evaluation of applications for new licenses and for evaluating compliance with Appendix G to 10 CFR Part 50.

REFERENCES

1. American Society for Testing and Materials, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM E 185-79, July 1979.[1]

2. American Society for Testing and Materials, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactors," ASTM E 185-82, July 1982.[1]

3. American Society of Mechanical Engineers, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the *ASME Boiler and Pressure Vessel Code*, New York, through 1988 Addenda and 1989 Edition.[2]

4. T.L. Dickson, "Generic Analyses for Evaluation of Low Charpy Upper-Shelf Energy Effects on Safety Margins Against Fracture of Reactor Pressure Vessel Materials," USNRC, NUREG/CR-6023, July 1993.[3]

5. R. Johnson, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," USNRC, NUREG-0744, Volume 1 (Revision 1) and Volume 2 (Revision 1), October 1982.[3]

6. Generic Letter No. 82-26, "NUREG-0744 Rev. 1; Pressure Vessel Material Fracture Toughness," Issued by Darrel G. Eisenhut, Director, Division of Licensing, NRR, USNRC, November 12, 1982.[4]

7. Letter from Warren H. Bamford, Chairman of the ASME Subgroup on Evaluation Standards for ASME Section XI, to James E. Richardson, USNRC, Subject: Response to NRC Request, A-11 Issue, February 20, 1991.[4]

8. American Society of Mechanical Engineers, "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels," Appendix K, A93, pp. 482.1-482.15, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1992 Edition, 1993 Addenda, New York, December 1993.[2]

9. American Society for Testing and Materials, "Standard Test Method for Determining J-R Curves," ASTM E 1152-87, May 1987.[1]

10. E.D. Eason, J.E. Wright, and E.E. Nelson, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," USNRC, NUREG/CR-5729, May 1991.[3]

11. A.W. Serkiz, "ASME Code Class 1, 2, and 3 Components, Components Supports, and Core Structures," Revision 1 to Appendix A to Section 3.9.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," pages 3.9.3-12 to 3.9.3-20, April 1984.[3]

12. American Society of Mechanical Engineers, Section III, "Nuclear Power Plant Components," of the *ASME Boiler and Pressure Vessel Code*, New York, through 1988 Addenda and 1989 Edition.[2]

13. J.M. Bloom, "Validation of the Deformation Plasticity Failure Assessment Diagram (DPFAD) Approach - The Case of An Axial Flaw in a Pressurized Cylinder," Transactions of the ASME, *Journal of Pressure Vessel Technology*, Volume 112, pp. 213-217, 1990.

14. USNRC, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.[4]

15. A.L. Hiser and J.B. Terrell, "Size Effects on J-R Curves for A-302B Plate," USNRC, NUREG/CR-5265, January 1989.[3]

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[1] Copies may be obtained from the American Society for Testing and Materials, 1916 Race Street, Philadelphia, PA 19103.

[2] Copies may be obtained from the American Society of Mechanical Engineers, 345 East 47th Street, New York, NY 10017.

[3] Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington DC 20555; telephone (202) 634-3273, fax (202) 634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082 (telephone (202) 512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

[4] Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington DC 20555; telephone (202)634-3273, fax (202)634-3343.

16. T.J. Griesbach and E. Smith, "A Review of the ASME Low Upper Shelf Toughness Evaluation Procedures for Nuclear Reactor Pressure Vessels," *Nuclear Engineering and Design*, Volume 130, No. 3, pp. 259-266, 1991.

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APPENDIX A

EXAMPLES

Several cases are provided here to demonstrate examples of the methods of analysis described in this regulatory guide.

Example 1 (Levels A&B Loading, PWR Vessel)

Consider the following geometric and material properties:

Vessel Geometry and Loading Conditions:
Vessel internal radius, $R_i = 86.5$ in.; A-533B vessel with generic welds
Base metal thickness, $t = t_{BM} = 8.444$ in.; Cladding thickness, $t_{cL} = 5/32$ in.
Total thickness, $t' = (t_{BM} + t_{cL}) = 8.6$ in.; Ratio $(R_i/t') = 10.06$
System accumulation pressure, $p_a = 2.75$ ksi; Cooldown transient = 100°F/hr

Base Metal Thermo-Elastic Properties:
Modulus of elasticity, E = 27E3 ksi; Poisson's ratio, $\nu = 0.3$
Yield stress, $\sigma_y = 80$ ksi; Ultimate stress, $\sigma_u = 90$ ksi
Flow stress, $\sigma_f = 85$ ksi; Fluid heat transfer coeff. = 1000 BTU/hr-ft²-°F
Thermal diffusivity = 0.98 in²/minute; $(E.\alpha)/(1 - \nu) = 0.305$ ksi/°F

Cladding Thermo-Elastic Properties:
Thermal expansion coefficient, $\alpha = 9.1E\text{-}6/°F$; Poisson's ratio, $\nu = 0.3$
Modulus of elasticity, E = 27E3 ksi; Thermal conductivity = 10 BTU/hr-ft-°F
Stress-free temperature of cladding = 550°F; Initial operating temp. = 550°F

The VISA-II code,[1] with modifications for printing K_{Ip}, K_{It}, and K_{cL} for 6-to-1 aspect ratio flaws, was used to perform analyses for determining transient thermo-mechanical stresses and temperature gradients across vessel wall thickness. An axial flaw with an aspect ratio of 6 to 1 was postulated to exist in the vessel internal wall. To account for the effect of crack-face pressure on stress intensity factor solutions in VISA-II, the accumulation pressure was adjusted to be equal to $[p.t'.\{1 + R_i/t'\}/R_i]$, 3.02 ksi. At a fixed crack depth of (0.25t'+0.1) inch, the temperature history prediction is shown in Figure A-1 for a transient with a constant cooldown rate of 100°F/hr.

With a factor of safety, SF, of 1.15 on accumulation pressure for Equation 1 of this guide, the applied J-integral history at a crack depth of (0.25t'+ 0.1) inch for mechanical and thermal stresses, including the cladding effects, is shown in Figure A-2. The applied J-integral reaches the peak steady-state value of 486 in.-lb/in.² in about 150 minutes. Also shown in Figure A-2 are the J-R curves for generic welds (Equations 17, 24-25) at three Charpy V-notch upper-shelf energy (CVN) values. These J-R curves were drawn for a crack extension, Δa, of 0.1 inch and for the temperature history, in Figure A-1, at a crack depth of (0.25t'+0.1) inch. A study of Figure A-2 shows an interesting trend that the crack initiation is predicted to take place at about 45 minutes into the transient (with crack-tip temperature of 500°F) where the applied-J value (= 445 in.-lb/in.²) is less than the peak steady-state value and is just equal to the material's J-R curve at CVN value of 40 ft-lb. Thus, the more detailed analysis results in a lower CVN value that satisfies the acceptance criteria.

In order to satisfy Equation 2, with a safety factor of 1.25 on accumulation pressure, Figure A-3 shows that CVN value should be greater than or equal to 41 ft-lb. This is significantly lower than the 47 ft-lb value obtained by using the steady-state applied J-integral approach for analyzing transients with constant cooldown rates.

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[1] F.A. Simonen et al., "VISA-II - A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure," USNRC, NUREG/CR-4486, March 1986.

Example 2 (Levels C and D Loading, PWR Vessel)

The problem statement was presented in a meeting of the ASME Section XI Working Groups on Flaw Evaluation and Operating Plant Criteria (in Louisville, Kentucky, on December 1, 1992), where results of the analyses were compared by the participants. The vessel geometry and material properties are:

PWR vessel internal radius, $R_i = 90.0$ inch; A-533B plate material thickness, $t = t_{BM} = 9.0$ inch; Cladding thickness, $t_{cL} = 0$, $R_i/t = 10$ Copper, Cu = 0.35 wt%; Nickel, Ni = 0.3 wt%; Initial $RT_{NDT} = 0.0°F$
Pre-irradiated $CVN_p = 108$ ft-lb (L-T orientation)
Surface fluence, $\phi t = 3.0E19$ n/cm²
Flaw orientation = Axial, in plate material; Flaw aspect ratio = 6 to 1
Fluid temperature at vessel surface, $T(tm) = [550 - 250\{1 - \exp(-0.1\ tm)\}]$°F with time, tm, in minutes.
Heat transfer coeff. = 320 BTU/hr-ft²-°F; Thermal diffusivity = 0.98 in.²/min
Elastic modulus, E = 28E3 ksi; Poisson's ratio, $\nu = 0.3$; $\alpha = 8.1E\text{-}6$ in./in.-°F
Yield stress, $\sigma_y = 80$ ksi; Flow stress, $\sigma_f = 85$ ksi

J-R curve: $J = (SF).[C1.(\Delta a)^{C2}.\exp\{C3.(\Delta a)^{C4}\}]$ in.-kip/in.²
where:

$\ln(C1) = [-2.89 + 1.22\ln(CVN_p) - 0.0027\,T + 0.014\,(\phi t)]$
C2 $= [0.077 + 0.116\ln(C1)]$
C3 $= [-0.0812 - 0.0092\ln(C1)]$
C4 $= -0.417$
SF $= 0.741$ for Level C events

The VISA-II code was used to determine thermal stress and temperature history for the Level C transient specified in the problem. It was found that at time tm = 20 minutes, the peak thermal stresses occur. The corresponding peak thermal stress intensity factor as a function of crack depth to vessel thickness ratio, a/t, of semi-elliptical flaws is given as:

$$K_{It} = [21.026+374.22(a/t)-1593.56(a/t)^2+2912.1(a/t)^3-2029.7(a/t)^4]\ \text{ksi}\sqrt{\text{in.}}\ \text{with}\ 0.05 \le a/t \le 0.5$$

Therefore, at a = 1 inch, $K_{It} = 46.6$ ksi$\sqrt{\text{in.}}$ At an internal pressure, p = 1 ksi, the pressure induced $K_{Ip} = 18.9$ ksi$\sqrt{\text{in.}}$ Now, if the pressure, p, is increased, then at a pressure of 6.75 ksi, the J-applied at $a = (0.1t + t_{cL} + 0.1)$ inch becomes equal to the material's J-R curve as shown in Figure A-4. This will mark an "initiation" of ductile flaw growth. The temperature at the crack-tip ($a = 0.1t + t_{cL}$) for time tm = 20 minutes is 400°F. If internal pressure p is further increased, in Figure A-4 it can be seen that at pressure p = 7.56 ksi the crack growth becomes unstable. That is, the slope of the J-applied curve becomes greater than the slope of the material's J-R curve.

Example 3 (Levels C and D Loading, BWR Vessel)

The problem statement is the same as in Example 2, except for a BWR vessel geometry. The vessel geometric details are:

BWR vessel internal radius, $R_i = 120.0$ inch; A-533B plate material
Thickness, $t = t_{BM} = 6.0$ in.; Cladding thickness, $t_{cL} = 0$; $R_i/t = 20$
Flaw orientation = Axial, in plate material; Flaw aspect ratio = 6 to 1

The VISA-II code was used to determine thermal stress and temperature history for the Level C transient specified in the problem. It was found that at time tm = 16 minutes, peak thermal stresses occur. The corresponding peak thermal stress intensity factor as a function of crack depth to vessel thickness ratio, a/t, of semi-elliptical flaws is given as:

$$K_{It} = [12.243+227.94(a/t)-972.71(a/t)^2+1785.2(a/t)^3-1249.3(a/t)^4]\ \text{ksi}\sqrt{\text{in.}},\ \text{with}\ 0.05 \le a/t \le 0.5$$

Therefore, at a = 1 inch, K_{lt} = 27.9 ksi$\sqrt{in}$. At an internal pressure, p = 1 ksi, the pressure-induced K_{Ip} = 37.0 ksi$\sqrt{in}$. If the pressure, p, is increased, at a pressure of 4.55 ksi, the J-applied at a = $(0.1t + t_{cL} + 0.1)$ inch becomes equal to the material's J-R curve as shown in Figure A-5, which will mark an "initiation" of ductile flaw growth. The temperature at the crack tip (a = $0.1t + t_{cL}$) for time tm = 16 minutes is 405°F. If the pressure, p, is further increased (see Figure A-5), it can be seen that at a pressure p = 4.75 ksi the crack growth has become unstable. The slope of the J-applied curve is now greater than the slope of the material's J-R curve.

Example 4 (Thermal K_{It} for Prescribed Levels C and D Loading, PWR Vessel)

For a PWR vessel, thermal K_{It} values are determined for a few prescribed cooldown rate (CR) transients. The geometric and material properties are given as:

Vessel Geometry and Loading Conditions:
Vessel internal radius, R_i = 86.875 in.; A-533B plate material with cladding
Base metal thickness, t = t_{BM} = 8.625 in.; Cladding thickness, t_{cL} = 5/16 in.
Total thickness, t' = $(t_{BM} + t_{cL})$ = 8.9375 in.; Ratio, (R_i/t') = 9.72
Thermal cooldown rate, CR = 100°F/hr to 600°F/hr (constant, for each analysis)
Inner wall temperature, $T_{initial}(R = R_i)$ = 550°F; $T_{final}(R = R_i)$ = 150°F

Base Metal Thermo-Elastic Properties:
Modulus of elasticity, E = 27E3 ksi; Poisson's ratio, ν = 0.3
Fluid-film heat transfer coefficient = 1000 BTU/hr-ft²-°F
Thermal diffusivity = 0.98 in²/minute; $(E\alpha)/(1 - \nu)$ = 0.305

Cladding Thermo-Elastic Properties:
Thermal expansion coefficient, α = 9.1E-6/°F; Poisson's ratio, ν = 0.3
Modulus of elasticity, E = 27E3 ksi; Thermal conductivity = 10 BTU/hr-ft-°F
Stress-free temperature of cladding = 550°F; Initial operating temp. = 550°F

The VISA-II code was used to determine temperature and thermal stress history for constant CR transients of 100°F/hr, 150°F/hr, 200°F/hr, 300°F/hr, 400°F/hr, 500°F/hr, and 600°F/hr. The corresponding peak thermal stress intensity factors, K_{It}, as a function of crack depth to vessel thickness ratio, a/t', for 6-to-1 aspect ratio semi-elliptical flaws, were computed using the VISA-II code. These are shown in Figure A-6 and are presented here in polynomial expressions using least-square fits as:

<u>For CR = 100°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [27.284 - 5.838\,(a/t') - 0.3548\,(a/t')^2 - 8.3858\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

<u>For CR = 150°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [32.003 + 40.012\,(a/t') - 138.2\,(a/t')^2 - 113.98\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

<u>For CR = 200°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [36.362 + 82.011\,(a/t') - 265.01\,(a/t')^2 + 226.9\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

<u>For CR = 300°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [43.667 + 150.77\,(a/t') - 474.9\,(a/t')^2 + 415.01\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

<u>For CR = 400°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [49.254 + 201.12\,(a/t') - 632.1\,(a/t')^2 + 557.87\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

<u>For CR = 500°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [53.552 + 237.64\,(a/t') - 749.6\,(a/t')^2 + 666.62\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

<u>For CR = 600°F/hr,</u> with $0.05 \le (a/t') \le 0.5$:
$$K_{It} = [56.927 + 264.21\,(a/t') - 838.6\,(a/t')^2 + 750.88\,(a/t')^3]\ \text{ksi}\sqrt{\text{in}}.$$

These results were also used in developing the unified Equation 13 for K_{I} where the constant CR and the normalized crack depth, a/t', are used as dependent variables. A least-squares statistical fit was performed to obtain Equation 13. The cross-product term, (CR)(a/t'), was also used in developing this fit, in addition to the polynomial terms in a/t' and CR.

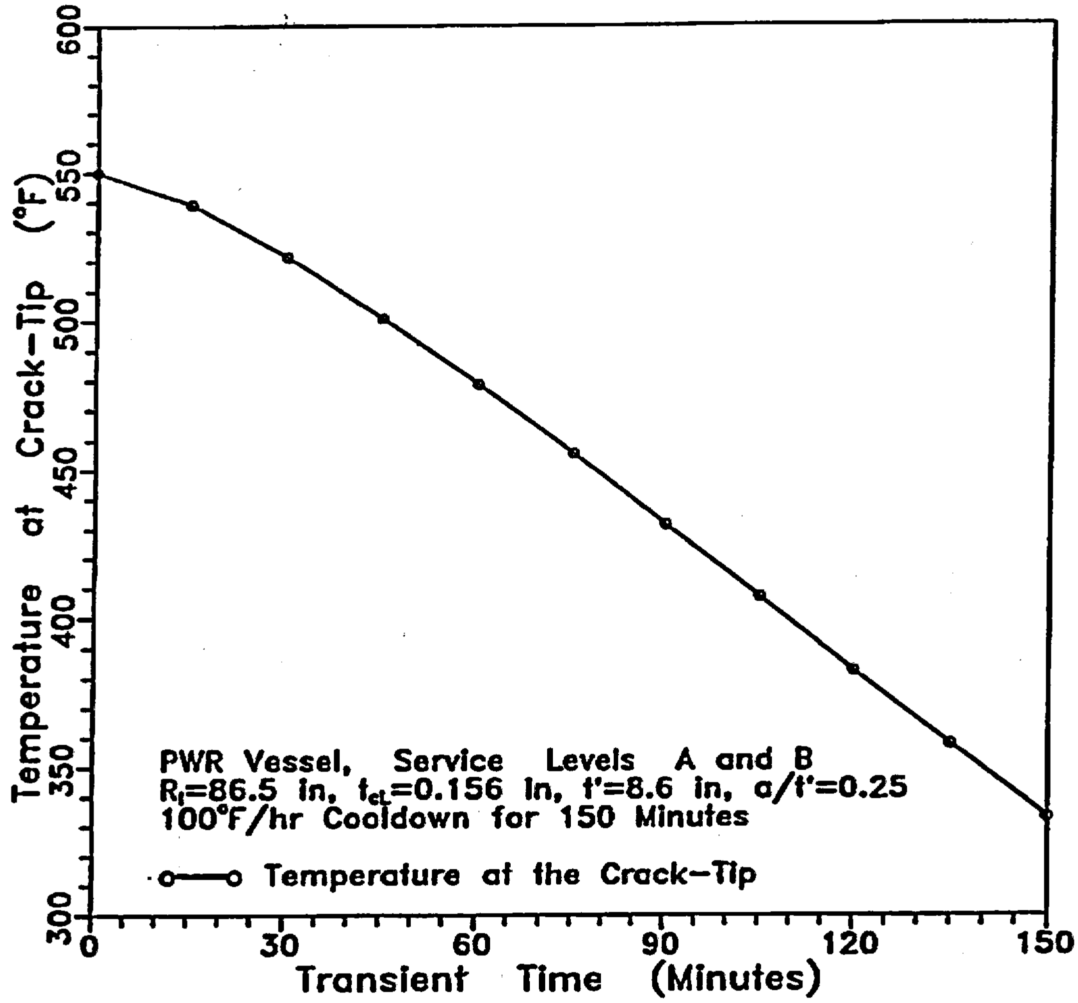

Figure A-1: Transient Temperature History at Crack Tip for Service Levels A and B.

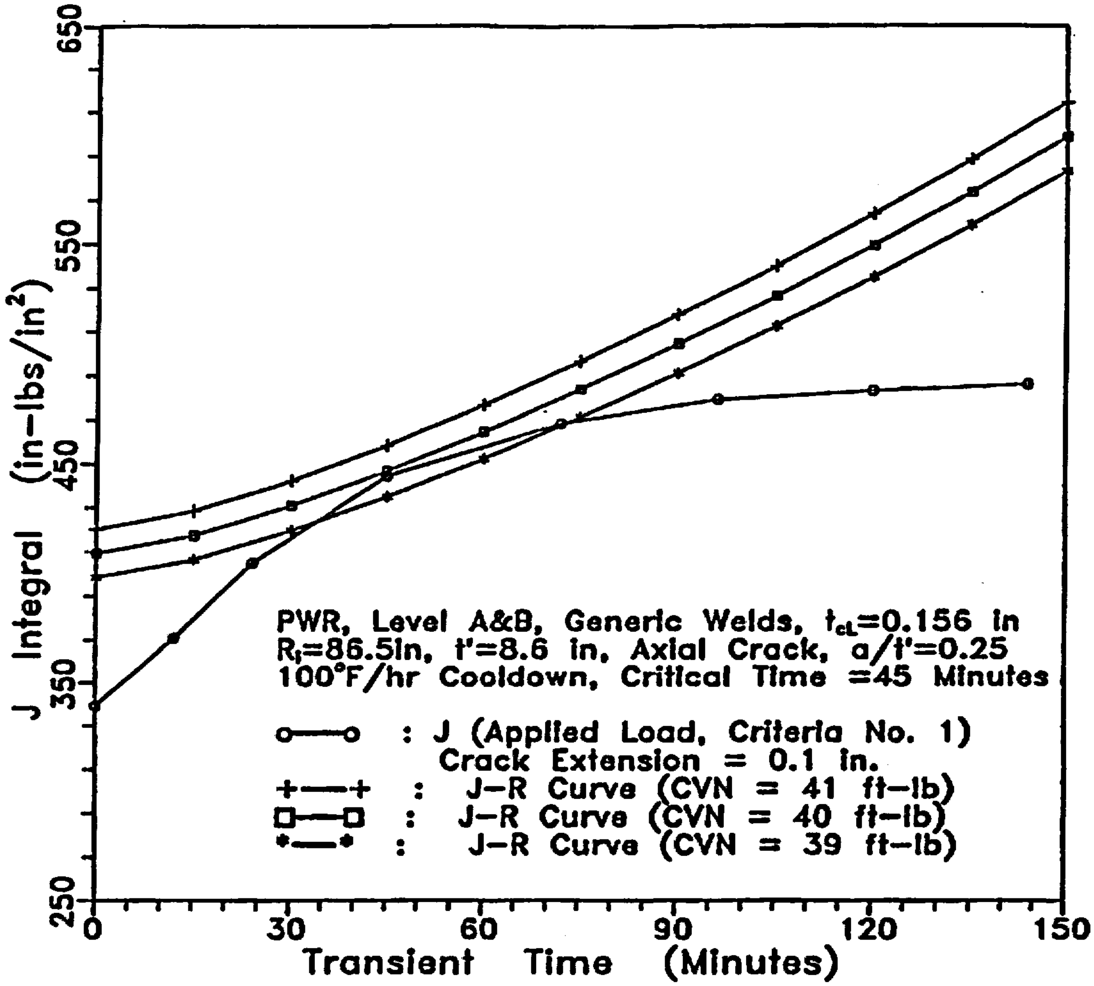

Figure A-2: J-Applied and J-R Curve History at Crack Tip for Service Levels A and B at a Crack Extension of 0.1 inch.

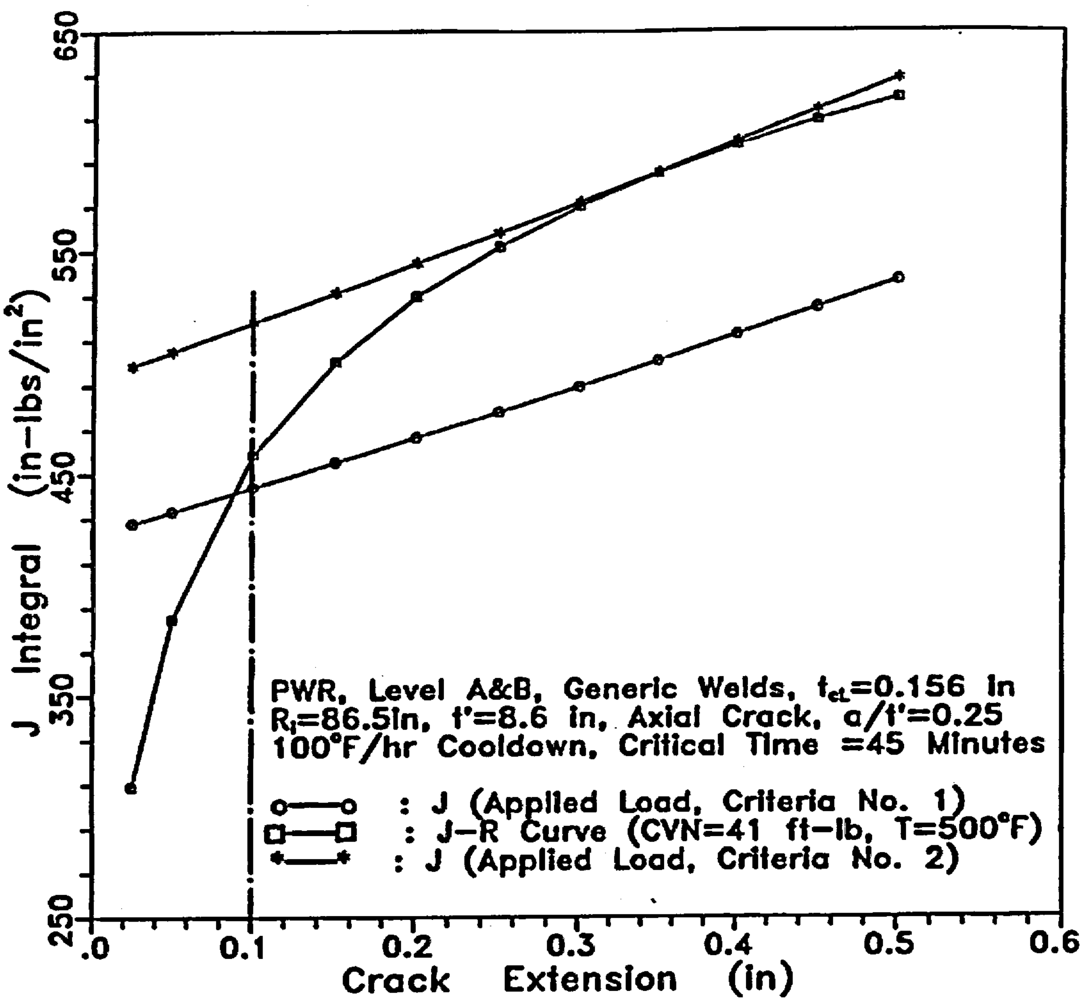

Figure A-3: Acceptable Upper-Shelf Energy in a PWR Vessel for Service Levels A and B.

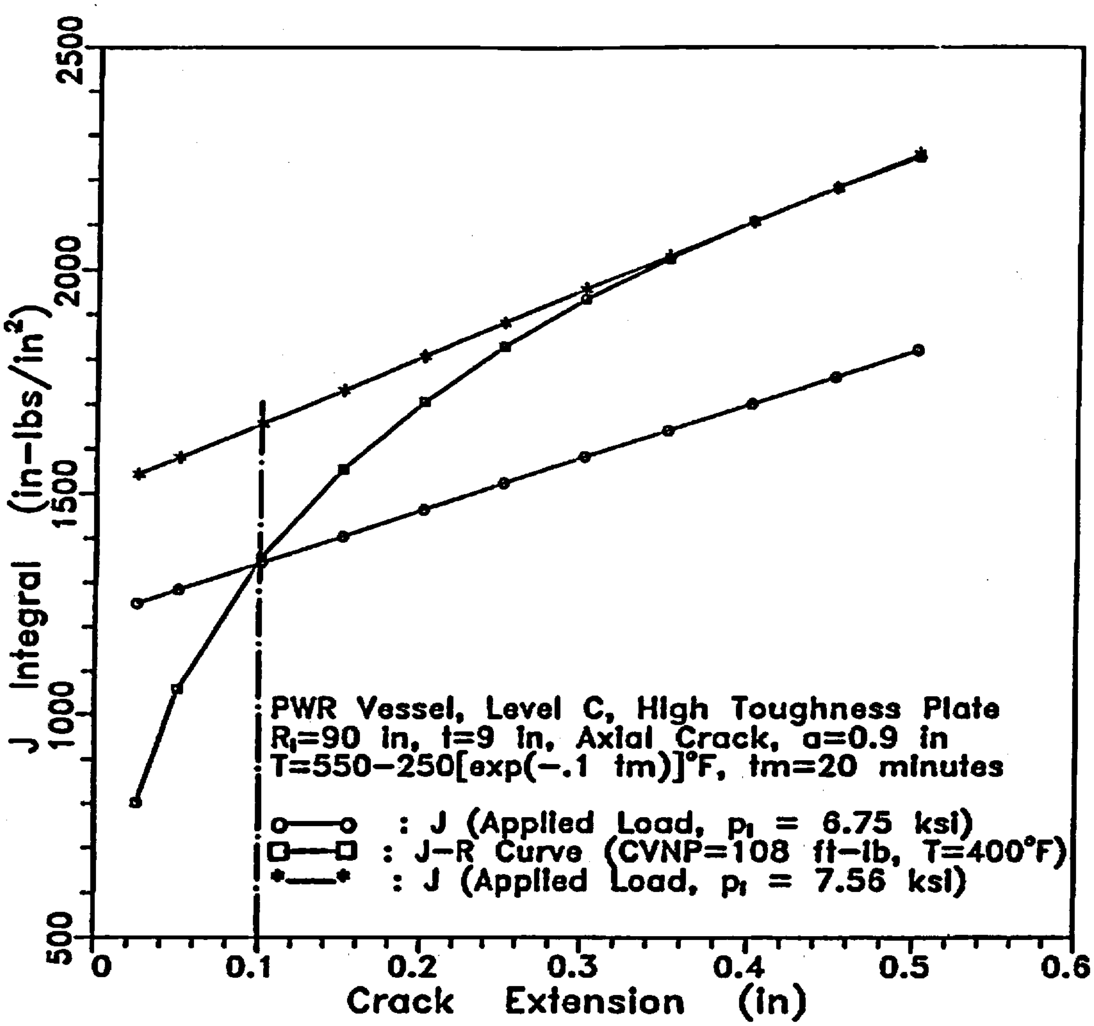

Figure A-4: Safety Margin Evaluation in a PWR Vessel for Service Level C.

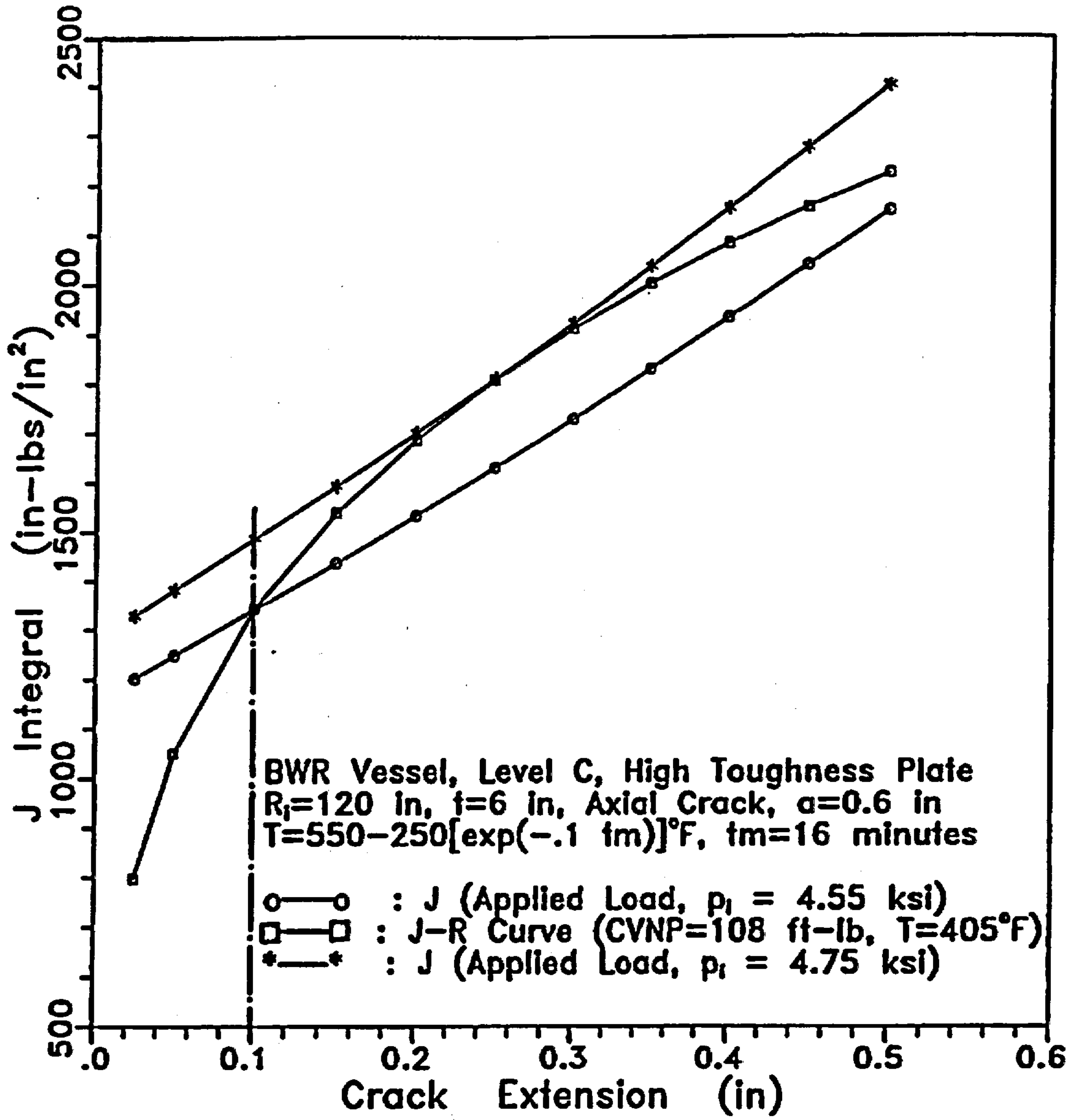

Figure A-5: Safety Margin Evaluation in a BWR Vessel for Service Level C.

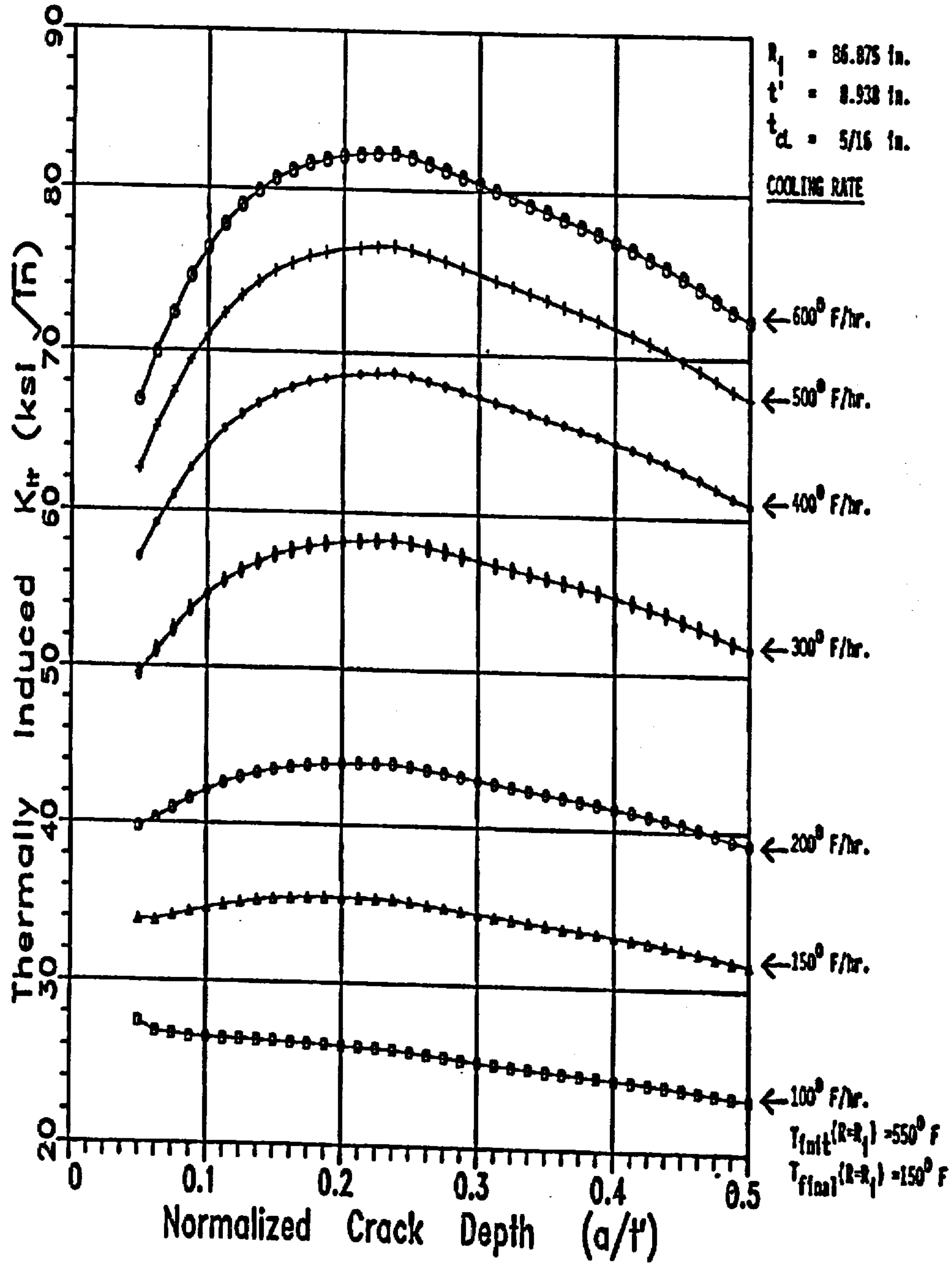

Figure A-6: Peak Thermal Stress Intensity Factors in a PWR Vessel for Transients with Several Different, but Constant, Cooldown Rates.

APPENDIX B

COMPUTATION OF STRESS INTENSITY FACTORS

Information about computing transient temperature gradient across the vessel wall thickness, thermal stresses, pressure, and thermal stress intensity factors (K_{Ip}, K_{It}) are provided in this Appendix as FORTRAN subroutines from the VISA-II code. Additional details on the computational method, theory used, limitations, and names of the major variables used are available in NUREG/CR-4486[1] and NUREG/CR-3384.[1] The computer code provided in this Appendix is for general illustration only, to show how the cladding effects could be incorporated for thermal stresses and thermal stress intensity factors caused by differential thermal expansion between the cladding and the base metal. Licensees should ensure that the computer codes they use include an indepth evaluation of these effects.

A description of cladding-induced thermal stress intensity factors is presented in Appendix A to NUREG/CR-4486. Limitations of the stress intensity factor correction factors for finite length semi-elliptical surface flaws are indicated in Appendix C to NUREG/CR-4486. In developing these correction factors, only uniform membrane and linear bending stresses were considered. In addition, the correction factors for circumferential flaws were assumed to be the same as the ones for axial flaws. Improved solutions may be used on a case-by-case basis if justified.

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[1] F.A. Simonen et al., "VISA-II - A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure," USNRC, NUREG/CR-4486, March 1986. D.L. Stevens et al., "VISA - A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure," USNRC, NUREG/CR-3384, September 1983. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. Copies of NUREG/CRs may be purchased at current rates from the U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

Taken From: VISA-II Code [NUREG/CR-4486 (1986), NUREG/CR-3384 (1983)]

```
C*******************************************************************************
SUBROUTINE SPKI
C*******************************************************************************C
Calculate Pressure Values, and, Stress Intensity Factor, PKI
   DIMENSION CONST(5)
   REAL I(5), IC(5)
   INTEGER CRACK, TIME
C   DETERMINE POLYNOMIAL REPRESENTATION OF PRESSURE
   CONST(1) = PDATA(1)
   CONST(2) = ((-25)*PDATA(1)+48*PDATA(2)-36*PDATA(3)+
   1          16*PDATA(4)-3*PDATA(5))/(3*TMAX)
   CONST(3) = (35*PDATA(1)-104*PDATA(2)+114*PDATA(3)-
   1          56*PDATA(4)+11*PDATA(5))*2/(3*TMAX**2)
   CONST(4) = ((-5)*PDATA(1)+18*PDATA(2)-24*PDATA(3)+
   1          14*PDATA(4)-3*PDATA(5))*16/(3*TMAX**3)
   CONST(5) = (PDATA(1)-4*PDATA(2)+6*PDATA(3)-4*PDATA(4)+
   1          PDATA(5))*32/(3*TMAX**4)
C  Calculate PRESSURE Component of Applied K, PKI, For Each Time & Crack Depth
   OUTRAD = RAD + TH
   FACTOR = RAD**2.0 / (OUTRAD**2.0 - RAD**2.0)
C
   DO 120  TIME = 1, 10
   TT = TMAX*TIME/10.0
   DO 110  CRACK = 1, ICMAX
   X = Z(CRACK)/TH
C  CALCULATE  INFLUENCE  COEFFICIENTS
   DO 100 M = 1, 5
    I(M)  = ZZ(M,1) + X*ZZ(M,2) + (X**2)*ZZ(M,3) + (X**3)*ZZ(M,4)
    IC(M) = ZZC(M,1) + X*ZZC(M,2) + (X**2)*ZZC(M,3) + (X**3)*ZZC(M,4)
 100 CONTINUE
   PRES(TIME) = CONST(1)+CONST(2)*TT+CONST(3)*TT**2+CONST(4)*TT*
   1            *3+CONST(5)*TT**4
   PKI(CRACK,TIME) = PRES(TIME)*((3.1416*Z(CRACK))**.5)*(10.5238*I(1)
   1            -1.1524*I(2)*X+0.1729*I(3)*(X**2)-0.0230*I(4)
   2            *(X**3)+0.0029*I(5)*(X**4))
```

```
      PKIC(CRACK,TIME) = 5*PRES(TIME)*((3.1416*Z(CRACK))**.5)*IC(1)
      RATIO = RAD / (10.0*TH)
      PKI(CRACK,TIME) = RATIO * PKI(CRACK,TIME)
      PKIC(CRACK,TIME) = RATIO * PKIC(CRACK,TIME)
C     CALCULATE HOOP STRESS
      SHOOP(CRACK,TIME) = FACTOR * PRES(TIME) *
     1           (1.0 + (OUTRAD/(RAD + Z(CRACK)))**2.0 )
  110 CONTINUE
C     CALCULATE LONGITUDINAL STRESS
      SLONG(TIME) =  PRES(TIME) * FACTOR
  120 CONTINUE
      RETURN
      END
C*********************************************************************************
      SUBROUTINE  TPOLY
C*********************************************************************************
C   CALCULATE WATER TEMPERATURES USING A  "POLYNOMIAL"  MODEL
              REAL TEMP(5), CONST(5), S(5), AN(4), Y(4,5), KTEST
      REAL K, KO, CP(4), SUM(4)
      INTEGER TIME, CRACK, CONSTK, CONSTE
      INTEGER Q
C   "POLYNOMIAL" Modeling of The Water Temperature
C   Determine Metal Temperature For EACH CRACK DEPTH  AND TIME INTERVAL
      DO 100 N = 1, 5
      TEMP(N) = TDATA(N) - TINT
  100 CONTINUE
C   FIT A "POLYNOMIAL" TO THE WATER TEMERATURE
      CONST(1) = TEMP(1)
      CONST(2) = ((-25)*TEMP(1) + 48*TEMP(2) - 36*TEMP(3) +
     1           16*TEMP(4) - 3*TEMP(5))/(3*TMAX)
      CONST(3) = (35*TEMP(1) - 104*TEMP(2) + 114*TEMP(3) -
     1           56*TEMP(4) + 11*TEMP(5))*2/(3*TMAX**2)
      CONST(4) = ((-5)*TEMP(1) + 18*TEMP(2) - 24*TEMP(3) +
     1           14*TEMP(4) - 3*TEMP(5))*16/(3*TMAX**3)
      CONST(5) = (TEMP(1) - 4*TEMP(2) + 6*TEMP(3) - 4*TEMP(4) +
     1           TEMP(5))*32/(3*TMAX**4)
      DO 150 TIME = 1, 10
```

```
      TT = TMAX*TIME/10.
C     EQUATION FOR THE TEMPERATURE OF THE WATER
      TWATER(TIME) = TINT+CONST(1)+ CONST(2)*TT + CONST(3)*TT**2 +
     1          CONST(4)*TT**3 + CONST(5)*TT**4
      DO 150 CRACK = 1, 5
      K = KO
  110 X = ZQ(CRACK)/TH
      TAU = K*TT/TH**2
      DO 120 M = 1, 5
       S(M) = CONST(M) * ((TH**2/K)**(M-1))
  120 CONTINUE
      DO 130 N = 1, 4
      ALNQ  = AL(N,Q)
      AN(N) = 2 * SIN(ALNQ)/(ALNQ + SIN(ALNQ)* COS(ALNQ))
      CP(N) = COS(ALNQ * (1-X))
      Y(N,1) = 1 - EXP(-(ALNQ**2)*TAU)
      DO 130 M = 2, 5
       Y(N,M) = TAU**(M-1) - (Y (N,M-1)/ALNQ**2)*(M-1)
130   CONTINUE
      DO 140 N = 1, 4
       ALNQ = AL(N,Q)
       SUM(N) = AN(N) * CP(N) * (S(1) * EXP(-(ALNQ**2*TAU)) + S(2)
     1       * Y(N,1)/ALNQ**2 +2*S(3)* Y(N,2)/ALNQ**2 + 3 *S(4) * Y(N,3)
     2       /ALNQ**2 +4 *S(5)*Y(N,4)/ALNQ**2)
  140 CONTINUE
C     EQUATION FOR THE QUARTER POINT TEMPERATURES
      TQ(CRACK,TIME) = TWATER(TIME) - SUM(1) - SUM(2) - SUM(3) - SUM(4)
C     CONTROL FOR THE CONSTANT KAPPA OPTION
      IF (CONSTK .EQ. 1) GO TO  150
C     TEST FOR THE ACCURACY OF KAPPA FOR THE GIVEN METAL TEMPERATURE,
C      IF THE DESIRED ACCURACY IS NOT OBTAINED, ITERATE ON  KAPPA
C      FOR THIS CRACK DEPTH AND TIME.
      KTEST = 1.030 - (5.97E-7)*((T(CRACK,TIME))**2)
      IF ((ABS(KTEST-K)) .LE. 0.0001) GO TO  150
      K = KTEST
      GO TO  110
  150 CONTINUE
```

```
      RETURN
      END
C*************************************************************************************
      SUBROUTINE  TEXP
C*************************************************************************************
C  Calculate  WATER TEMPERATURES Using an  "Exponential Decay"  Model
      REAL B, KTEST, K, KO, SUM(4)
      INTEGER CRACK, TIME, CONSTK, CONSTE
      INTEGER Q
C  EXPONENTIAL DECAY MODEL OF THE WATER TEMPERATURE
      DO 130 TIME = 1, 10
      TT = TMAX*TIME/10.
C  EQUATION FOR THE TEMPERATURE OF WATER
      TWATER(TIME) = TO + DT * (1-EXP(-BE*TT))
      DO 130 CRACK = 1, 5
      K = KO
  100 WSQ = BE*TH*TH/K
      TAU = K*TT/(TH*TH)
      DO  120 N = 1, 4
       ALNQ = AL(N,Q)
       B  = -DT*((2*SIN(ALNQ)/(ALNQ+(SIN(ALNQ))*(COS(ALNQ))))
     1      *(EXP(-(ALNQ**2*TAU))-EXP(-WSQ*TAU))/((ALNQ**2/WSQ)-1))
       X  = ZQ(CRACK)/TH
       SUM(N) = B * COS(ALNQ*(1-X))
  120 CONTINUE
C  EQUATON FOR THE "QUARTER POINTS"  TEMPERATURE VALUES
      TQ(CRACK,TIME) = TWATER(TIME) - SUM(1) - SUM(2) - SUM(3) - SUM(4)
C  CONTROL FOR THE CONSTANT KAPPA OPTION
      IF (CONSTK .EQ. 1) GO TO  130
C  TEST FOR KAPPA ACCURACY AND CONTROL OF KAPPA OPTION
      KTEST = 1.030 - (5.97E-7)*((T(CRACK,TIME))**2)
      IF ((ABS(KTEST-K)) .LE. 0.0001) GO TO  130
      K = KTEST
      GO TO  100
  130 CONTINUE
      RETURN
      END
```

```
C*****************************************************************************
   SUBROUTINE SKIT
C*****************************************************************************
C  Calculate Stress and Temperature at Crack-Tip and  Thermal Stress
C  Intensity Factor,  SKIt
   REAL E(5,10), CC(5), I(5), IC(5)
   INTEGER CRACK, TIME
   INTEGER Q, CONSTE, CONSTK
C  DETERMINE POLYNOMIAL REPRESENTATION OF TEMPERATURE PROFILE
C  CONVERT CLAD THERMAL CONDUCTIVITY TO INCH AND MINUTE UNITS
   CCOND = CCOND / (12.0*60.0)
   COND  = COND / (12.0*60.0)
   DO 105 TIME = 1, 10
    TQ1 = TQ(1,TIME)
    TQ2 = TQ(2,TIME)
    TQ3 = TQ(3,TIME)
    TQ4 = TQ(4,TIME)
    TQ5 = TQ(5,TIME)
    C1 = TQ1
    C2 = (-25*TQ1+48*TQ2-36*TQ3+16*TQ4-3*TQ5)/(3*TH)
    C3 = (35*TQ1-104*TQ2+114*TQ3-56*TQ4+11*TQ5)*(2.0/3.0*TH**(-2))
    C4 = (-5*TQ1+18*TQ2-24*TQ3+14*TQ4-3*TQ5)*(16.0/3.0*TH**(-3))
    C5 = (TQ1-4*TQ2+6*TQ3-4*TQ4+TQ5)*(32.0/3.0*TH**(-4))
C  CALCUATE TEMPRATURE AT THE CRACK TIPS
    DO 100 CRACK = 1, ICMAX
     T(CRACK,TIME) = C1+C2*Z(CRACK)+C3*(Z(CRACK)**2)
  1            +C4*(Z(CRACK)**3)+C5*(Z(CRACK)**4)
 100 CONTINUE
   IF (CTH .LE. 0.0) GO TO  105
   T(1,TIME) = T(2,TIME) - (COND/CCOND)*(T(2,TIME)-T(1,TIME))
105  CONTINUE
   IF (CONSTE .EQ. 1) GO TO  120
   DO 110 TIME = 1, 10
   DO 110 CRACK = 1, 5
    E(CRACK,TIME) = 0.286+(5.400E-5 * (TQ(CRACK,TIME)))
  1          -(2.600E-8 * (TQ(CRACK,TIME))**2)
 110 CONTINUE
```

```
      GO TO  140
 120 DO 130  TIME  = 1, 10
     DO 130  CRACK = 1, 5
      E(CRACK,TIME) = EDATA
 130 CONTINUE
C  DETERMINE POLYNOMIAL REPRESENTATION OF STRESS DIST
 140 DO 170 TIME = 1, 10
      DO 150 CRACK = 1, 5
       CC(CRACK) = E(CRACK,TIME)*TQ(CRACK,TIME)
 150 CONTINUE
      A1 = CC(1)
      A2 = (-25*CC(1)+48*CC(2)-36*CC(3)+16*CC(4)-3*CC(5))/3.0
      A3 = (35*CC(1)-104*CC(2)+114*CC(3)-56*CC(4)+11*CC(5))*(2.0/3.0)
      A4 = (-5*CC(1)+18*CC(2)-24*CC(3)+14*CC(4)-3*CC(5))*(16.0/3.0)
      A5 = (CC(1)-4*CC(2)+6*CC(3)-4*CC(4)+CC(5))*(32.0/3.0)
      SIG1 =  A2/2.0 + A3/3.0 + A4/4.0 + A5/5.0
      SIG2 = -A2
      SIG3 = -A3
      SIG4 = -A4
      SIG5 = -A5
C  CALCULATE STRESS AT CRACK TIPS
      DO 170 CRACK = 1, ICMAX
       X = Z(CRACK)/TH
       STRESS(CRACK,TIME) = SIG1 + SIG2*X + SIG3*(X**2)
    1              + SIG4*(X**3) + SIG5*(X**4)
C  CALCULATE INFLUENCE FUNCTIONS
       DO 160 M = 1, 5
        I(M)  = ZZ(M,1) + X*ZZ(M,2)+ (X**2)*ZZ(M,3)+ (X**3)*ZZ(M,4)
        IC(M) = ZZC(M,1)+X*ZZC(M,2)+(X**2)*ZZC(M,3)+(X**3)*ZZC(M,4)
 160 CONTINUE
       A = Z(CRACK)
C  EQUATION FOR THE THERMAL STRESS INTENSITY
       TK(CRACK,TIME)  = ((3.1416*A)**.5)*(SIG1*I(1)
    1              +SIG2*I(2)*X+SIG3*I(3)*X**2
    2              +SIG4*I(4)*X**3+SIG5*I(5)*X**4)
       TKC(CRACK,TIME) = ((3.1416*A)**.5)*(SIG1*IC(1)+SIG2*IC(2)
    1      *X+SIG3*IC(3)*X**2+SIG4*IC(4)*X**3+SIG5*IC(5)*X**4)
```

```
 170 CONTINUE
   RETURN
   END
C*************************************************************************
   SUBROUTINE  KICLAD
C*************************************************************************
C  THIS SUBROUTINE CALCULATES STRESSES AND STRESS INTENSITY FACTORS
C  DUE TO THE PRESENCE OF "CLADDING" ON THE I.D. SURFACE OF THE VESSEL
   INTEGER CRACK, TIME
   INTEGER CONSTE, CONSTK, Q
   REAL IO, I1
   DO 170  TIME = 1, 10
C  CALCULATE STRESS DISTRIBUTION THROUGH VESSEL WALL
C  TEMP AT CLAD/BASE METAL INTERFACE
   T1 = 0.5*(T(2,TIME) + T(3,TIME))
C  TEMPERATURE AT THE VESSEL I.D.
   TO = T(1,TIME)
C  STRESS-FREE TEMPERATURE
   TI = SFREET
C  CALCULATE STRESS DISTRIBUTION DUE TO CLAD
C  SIGC1 = STRESS IN CLAD AT VESSEL I.D.
C  SIGC2 = STRESS IN CLAD AT CLAD/BASE METAL INTERFACE
C  SIGB1 = STRESS IN BASE METAL AT CLAD/BASE METAL INTERFACE
C  SIGB2 = STRESS IN BASE METAL AT VESSEL O.D.
   DELEA = CLADE*CALPHA*(1-ARATIO)/(1-CLADNU)
C  CALCULATE STRESS IN CLAD (KSI)
   SIGC1 = DELEA * (TI - TO)
   SIGC2 = DELEA * (TI - T1)
C  CALCULATE FORCE DEVELOPED IN CLAD
   FCLAD = CTH*0.5*(SIGC1 + SIGC2)
C  CALCULATE STRESSES IN BASE METAL (KSI)
   RO = RAD
   R1 = RAD + CTH
   R2 = RAD + TH
   CONST = 1.0/((R2/R1)**2.0-1.0)*(RO-R1)/R1*DELEA
   1     *(TI-0.5*(TO+T1))
   SIGB1 = CONST * (1 + (R2/R1)**2.0)
```

```
      SIGB2 = CONST * 2.0
C     CALCULATE FORCE DEVELOPED IN BASE METAL
      FBASE = (CTH-TH)*0.5*(SIGB1+SIGB2)
C     ADJUST SIGB1 AND SIGB2 TO BALANCE FORCES FCLAD AND FBASE
      SIGINC = 0.5*(SIGB1-SIGB2)
      SIGAVE = 0.5*(SIGB1+SIGB2)*FCLAD/FBASE
      SIGB1 = SIGAVE + SIGINC
      SIGB2 = SIGAVE - SIGINC
C     CALCULATE CONSTANTS DESCRIBING STRESS DISTRIBUTION
C     QI = SLOPE OF CLAD STRESS DISTR.
      QI = (SIGC1-SIGC2)/SIGC1/(CTH/TH)
C     P = SLOPE OF BASE METAL STRESS DISTR.
      P = (SIGB2-SIGB1)/SIGC1 / ((TH-CTH)/TH)
C     -R = INTERCEPT OF BASE METAL STRESS GRAD. AT VESSEL I.D.
      R = -(SIGB1/SIGC1 - P*CTH/TH)
C     CALCULATE STRESS AND KI DUE TO CLAD FOR ALL Z(CRACK)'S
C     KI AT THE I.D. SURFACE EQUALS ZERO (I.E.,CRACKDEPTH = ZERO)
      SCLAD(1,TIME) = SIGC1
      CLADK(1,TIME) = 0.0
C     KI IN CLAD NEAR CLAD/BASE METAL INTERFACE
      SCLAD(2,TIME) = SIGC2
      ALP = Z(2)/TH
      IO = 1.122+0.9513*ALP-0.624*ALP**2.0+8.3306*ALP**3.0
      I1 = 0.6825+0.3704*ALP-0.0832*ALP**2.0+2.8251*ALP**3.0
      CLADK(2,TIME) = SQRT(3.14159*Z(2))*SIGC1*(IO-QI*ALP*I1)
C     CALCULATE KI IN BASE METAL
      XI = CTH/TH
      DO 170 CRACK = 3, 35
      ALP = Z(CRACK)/TH
      SCLAD(CRACK,TIME) = (-R+ALP*P)*SIGC1
      IO = 1.122+0.9513*ALP-0.624*ALP**2.0+8.3306*ALP**3.0
      CLADK(CRACK,TIME) = SQRT(3.14159*Z(CRACK))*SIGC1*1.751938
     1          *((IO-0.63662)*((1.0+R)*ASIN(XI/ALP)+ALP*((QI+R*P)
     2    *SQRT(1.-(XI/ALP)**2.)-QI)-1.570796*R)+(IO-1.0)*(((1.0+R)-XI/2.
     3    *(QI+R*P))*SQRT(1.-(XI/ALP)**2.)+ALP/2.0*(QI+R*P)*ASIN(XI/ALP)
     4    -1.0-0.7894*R*P*ALP))
  170 CONTINUE
```

```
      RETURN
      END
C*******************************************************************************
      SUBROUTINE  FACMB (AAA, BBB, THH, FMA, FMB, FBA, FBB)
C*******************************************************************************
C  THIS SUBROUTINE CORRECTS FOR "FINITE LENGTH" SEMI-ELLIPTICAL FLAWS
      DIMENSION ZM(2,4), ZB(2,4), Z(2)
      DIMENSION X1(12), YM(12,4), YB(12,4), Y(4)
      DATA X1/0., .0125, .025, .0375, .05, .075, .1,.15,.2,.3,.4,.5/
      DATA Y/ .05, .25, .5, .8 /
      DATA YM/ 1.0,.99,.98,.96,.95,.91,.87,.80,.75,.66,.60,.55,
     1       1.0,.94,.88,.83,.80,.76,.73,.68,.63,.55,.49,.44,
     2       1.0,.88,.77,.69,.64,.59,.55,.49,.44,.36,.31,.27,
     3       1.0,.72,.56,.48,.43,.38,.35,.29,.24,.18,.15,.13 /
      DATA YB/ 1.0,.98,.97,.95,.94,.92,.89,.85,.82,.74,.66,.58,
     2       1., .93,.88,.84,.80,.75,.72,.67,.63,.57,.50,.43,
     2       1., .84,.71,.63,.57,.49,.45,.39,.35,.29,.23,.18,
     3       1., .69,.50,.38,.29,.20,.14,.08,.05,.02,-.01,-.04/
      DATA Z/ 0.0, 0.5 /
      DATA ZM/ .44, .55, .40,.48,.31,.31,.23,.17 /
      DATA ZB/ .50, .62, .63, .67, .58, .50, .43,.32 /
      AOL = AAA/(2.0*BBB)
      AOT = AAA/THH
      DO 100 I = 1, 3
       J = I
       IF( Y(I+1) .GT. AOT )  GO TO  110
  100 CONTINUE
  110 N1 = J
      N2 = J+1
      DO 120 I = 1, 11
       J = I
       IF ( X1(I+1) .GT. AOL )  GO TO  130
  120 CONTINUE
  130 M1 = J
      M2 = J+1
      FAC1 = (AOL-X1(M1))/(X1(M2)-X1(M1))
      XX1  = YM(M1,N1)+FAC1*(YM(M2,N1)-YM(M1,N1))
```

```
XX2 = YM(M1,N2) + FAC1*(YM(M2,N2) - YM(M1,N2))
FAC = (AOT -Y(N1))/(Y(N2)-Y(N1))
IF (AOT .LT. 0.05 ) FAC = 0.0
IF ( AOT .GT. 0.80 ) FAC = 1.0
FMA = XX1 + FAC*( XX2 - XX1 )
XX1 = YB(M1,N1) + FAC1*(YB(M2,N1)-YB(M1,N1))
XX2 = YB(M1,N2) + FAC1*(YB(M2,N2)-YB(M1,N2))
FBA = XX1 + FAC*( XX2 - XX1 )
FAC1 = AOL/0.5
XX1 = ZM(1,N1) + FAC1*(ZM(2,N1)-ZM(1,N1))
XX2 = ZM(1,N2) + FAC1*(ZM(2,N2)-ZM(1,N2))
FMB = XX1 + FAC*( XX2-XX1)
XX1 = ZB(1,N1) + FAC1*(ZB(2,N1)- ZB(1,N1))
XX2 = ZB(1,N2) + FAC1*(ZB(2,N2)- ZB(1,N2))
FBB = XX1 + FAC*(XX2 - XX1)
RETURN
END
```

REGULATORY ANALYSIS

1. STATEMENT OF THE PROBLEM

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that the reactor vessel beltline materials ". . . must have Charpy upper-shelf energy of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." This Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lb," has been developed to provide acceptance criteria and analysis methods acceptable to the NRC staff for demonstrating margins equivalent to those in Appendix G to Section III of the ASME Code.

Publication of regulatory guidance was undertaken because no comprehensive guidance currently exists, and there are reactors, both pressurized water reactors and boiling water reactors, with upper-shelf energy that is projected to fall below the 50 ft-lb regulatory limit before the end of the current license period. Without comprehensive regulatory guidance, each affected licensee will have to submit a plant-specific analysis, including acceptance criteria and evaluation methods, and the staff will have to evaluate each submittal without the benefit of stated acceptance criteria and approved evaluation methods.

2. OBJECTIVES

The objective of this guide is to provide acceptance criteria and evaluation methods acceptable to the NRC staff for demonstrating margins equivalent to those in Appendix G to Section III of the ASME Code for those beltline materials whose Charpy upper-shelf energy falls below the regulatory limit provided in Appendix G to 10 CFR Part 50.

3. ALTERNATIVES

Two alternatives to issuing evaluation procedures for pressure vessels with Charpy upper-shelf energy less than 50 ft-lb were considered: (1) endorse actions being implemented by Section XI of the ASME Code and (2) take no action.

3.1 Endorse ASME Code, Section XI, Appendix K

The ASME, in Section XI, has published Appendix K[1] that provides acceptance criteria and evaluation procedures for pressure vessels with Charpy upper-shelf energy less than 50 ft-lb. However, the Appendix K evaluation procedures currently address only Service Levels A and B, and no guidance on specific materials properties is provided. It is important that all four service levels be considered in the evaluations, and it is important that specific guidance on estimating material properties be provided. Given the ASME codification process, and the process whereby the NRC endorses ASME appendices and code cases, the time delay in obtaining suitable guidance would be excessive. At present, the ASME's Appendix K does not provide complete guidance. As discussed above, Appendix K does not provide information on the selection of transients, and it gives very little detail on the selection of material properties. As such, a request for revision of Appendix K to Section XI of the ASME Code will have to be made.

3.2 Take No Action

As discussed in SECY-93-048,[2] "Status of Reactor Pressure Vessel Issues Including Compliance With 10 CFR Part 50, Appendices G and H," using the NRC staff's generic criteria for estimating Charpy upper-shelf energy, there are currently 15 plants that would have calculated upper-shelf energy less than 50 ft-lb and 3 others that would have upper-shelf energy below 50 ft-lb before the end of their operating licenses. Appendix G to 10 CFR Part 50 requires that licensees submit

[1] Appendix K (previously, Code Case N-512), "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels," American Society of Mechanical Engineers, Section XI, 1993.

[2] James M. Taylor, Executive Director for Operations, SECY-93-048, Policy Issue (Information) for the Commissioners, USNRC, February 25, 1993. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

analyses to demonstrate margins equivalent to those in Appendix G to Section III of the ASME Code 3 years before the upper-shelf energy of any beltline materials falls below 50 ft-lb. Therefore, taking no action is not a viable alternative.

4. COSTS AND BENEFITS OF ALTERNATIVES

The cost and benefits of the two alternatives discussed above are presented here.

4.1 Endorse Appendix K to ASME Code Section XI

The acceptance criteria proposed in Appendix K to ASME Section XI are identical to those proposed in this regulatory guide. The regulatory guide analysis procedures for Service Levels A and B were taken from Appendix K. However, the guide provides procedures applicable to Service Levels C and D. The regulatory guide provides specific guidance on appropriate material properties and on selection of transients for consideration, whereas Appendix K does not provide these procedures and guidance. Without this guidance, each affected licensee would have to develop appropriate procedures for Service Levels C and D, justify the choice of transients, and develop plant-specific material properties.

It is estimated that without the guidance of this regulatory guide, developing plant-specific procedures and material properties and applying them to check and report the analysis results would require an additional 6 staff-months (1040 hours) for each affected licensee. Assuming that half of the affected licensees either belong to owners' groups or could make use of common data, the total additional burden on the licensees that would be incurred by plant-specific analyses is estimated as 9 plants x 6 staff-months per plant, or 54 staff-months (9360 hours).

In addition to the increased burden on the licensees, it is estimated that an additional 1.5 NRC staff-month would be required to review each plant-specific submittal. Thus, the total increased burden on the NRC staff, assuming that half of the affected plants can be grouped, is estimated to be 9 plants x 1.5 staff-month per plant, or 13.5 staff-months (2340 hours). This estimate assumes that there would be only minor discussions with the licensees.

4.2 Take No Action

As discussed in Section 3.2 above, taking no action is judged to be a nonviable alternative.

5. DECISION RATIONALE

It is recommended that the regulatory guide be issued because it would offer a comprehensive set of acceptance criteria, evaluation procedures, and material properties that can be used to perform the analyses required under Appendix G to 10 CFR Part 50 for those pressure vessels that have Charpy upper-shelf energy of any beltline material that falls below 50 ft-lb. Issuing the regulatory guide is recommended over the alternative of endorsing Appendix K to ASME Section XI because Appendix K does not currently include (1) analysis procedures for Service Levels C and D, (2) guidance on selecting the transients for evaluation, or (3) details on temperature-dependent material properties. Further, it is estimated that preparing plant-specific analyses that include the procedures and data that are not addressed in Appendix K would require approximately 54 staff-months of effort for the industry and approximately 9 staff-months for the NRC to review the additional information.

The NRC staff considered the possibility of working with the ASME Code Section XI working group to modify Appendix K to include the missing procedures and data. However, given the number of plants that could need the guidance in the near term, and given the ASME codification process and the NRC's process for endorsing ASME documents, the time needed to modify and endorse Appendix K was judged to be excessive.

The efficacy of the procedures in the regulatory guide was demonstrated by generic bounding calculations[3] performed by the NRC staff in preparing SECY-93-048. These calculations demonstrated that the requirement in Appendix G to 10 CFR Part 50 to demonstrate margins equivalent to those in Appendix G to Section III to the ASME Code could be satisfied for materials with Charpy upper-shelf energy less than 50 ft-lb for all the generic vessel geometries and material combinations considered.

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[3] Charles Z. Serpan, Jr., NRC, Memorandum to Jack Strosnider, NRC, January 15, 1993, "Generic Bounding Analyses for Evaluation of Low Charpy Upper-Shelf Energy Effects on Safety Margins Against Fracture of RPV Beltline Plate and Weld Materials"; Charles Z. Serpan, Jr., NRC, Memorandum to Jack Strosnider, NRC, February 8, 1993, "Additional Information Regarding Results of Generic Bounding Analyses for Evaluation of Pressure Vessels Fabricated Using Low Charpy Upper-Shelf Energy Materials." Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

The regulatory guide acceptance criteria were taken directly from the ASME efforts. The criteria were developed by the ASME Code Section XI working group over an 11-year period and represent the collective judgment of a body of experts representing the NRC staff, research contractors, nuclear utilities, nuclear power plant vendors, consultants, and academia. Similarly, the evaluation procedures for Service Levels A and B were developed by this group. The procedures in the regulatory guide for Service Levels A and B are essentially identical to those in Appendix K to ASME Section XI. Thus, the acceptance criteria and the evaluation procedures for the service levels that generally control the analyses are based on the consensus technical opinion of a large group of technical experts and were developed over an extended period.

The evaluation procedures for Service Levels C and D were developed by the staff and build on the procedures for Service Levels A and B. As part of a continuing effort by the ASME Section XI working group, the NRC staff has compared the regulatory guide procedures to other procedures that are being developed by various organizations. The comparison was very favorable, with the procedures proposed in the regulatory guide predicting lower acceptable Charpy upper-shelf energy values than would be predicted by the other procedures, which were less rigorous and, consequently, more conservative.

The procedures for transient selection are based on procedures that have already been endorsed by the staff. Alternatively, generic bounding transients can be used if justified.

The guidance on material properties is based on a state-of-the-art statistical evaluation of all available fracture toughness data. A broad range of alternatives is offered in the regulatory guide so that methods acceptable to the staff are offered for virtually every situation and combination of circumstances.

The regulatory guide provides timely, cost-effective guidance that is based on the consensus of a large group of technical experts representing diverse backgrounds and interests. The specific guidance is comprehensive and would provide an effective and definitive approach to performing equivalent margin analyses.

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NUCLEAR REGULATORY COMMISSION
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Mr. John T. Conway
Executive Director
Joint Committee on Atomic Energy
Congress of the United States

Dear Mr. Conway:

Enclosed for the information of the Joint Committee on Atomic Energy are tentative "Supplementary Regulatory Criteria for ASME Code-Constructed Nuclear Pressure Vessels" dated August 14, 1967. These criteria have been developed by the AEC's Regulatory Staff with the cooperation of the Commission's Reactor Development and Technology Staff and national laboratories. The Regulatory Staff also worked closely with the ACRS in the development of the criteria, and they reflect ACRS review and comment. The criteria are an organized tabulation of requirements above those specified by the present ASME Nuclear Pressure Vessel Code which are now being imposed on a case-by-case basis by the ACRS and the Regulatory Staff, or which we think should be imposed for future cases. They reflect to a considerable degree current practice in design, fabrication, and inspection of pressure vessels for water-cooled power reactors, but are considered to be generally applicable to pressure vessels for other power reactors as well.

Over the past several months, industry code groups have made considerable progress in upgrading existing codes and standards applicable to nuclear pressure vessels, and they are actively considering further steps in this direction. We are making these criteria available to these groups and to others in the nuclear industry since they may be useful in these efforts. Pending further development, the supplementary criteria will provide interim guidance to the nuclear industry concerning AEC regulatory requirements.

23404

A copy of the public announcement to accompany the supplementary criteria when released also is enclosed.

Sincerely yours,

Harold L. Price

Harold L. Price
Director of Regulation

Enclosures:
1. "Supplementary Regulatory Criteria for ASME Code-Constructed Nuclear Pressure Vessels," August 14, 1967
2. Press Release

ATOMIC ENERGY COMMISSION

SUPPLEMENTARY REGULATORY CRITERIA

FOR ASME CODE-CONSTRUCTED NUCLEAR PRESSURE VESSELS

August 14, 1967

ATOMIC ENERGY COMMISSION

TENTATIVE

REGULATORY SUPPLEMENTARY CRITERIA

FOR ASME CODE-CONSTRUCTED NUCLEAR PRESSURE VESSELS

August 23, 1967

23430

ATOMIC ENERGY COMMISSION

SUPPLEMENTARY REGULATORY CRITERIA FOR ASME CODE-CONSTRUCTED NUCLEAR PRESSURE VESSELS*

*References to the "Code," "Code Paragraphs," or "Code Appendix" relate to the ASME Boiler and Pressure Vessel Code Section III, Rules for Construction of Nuclear Vessels, 1965 Edition, and the published addenda.

INTRODUCTION

Nuclear pressure vessels are among those components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety. In accordance with Criterion 1 of the proposed General Design Criteria for Nuclear Power Plant Construction Permits, which were published in the Federal Register (32 FR 10213) on July 11, 1967, these components must be designed and fabricated to quality standards reflecting the importance of the safety function which they perform.

For this reason, it has been the practice of the Commission to require on a case-by-case basis that nuclear pressure vessels be designed, fabricated, and inspected to quality standards which supplement those presently specified by industry codes. To formalize these requirements, the AEC's regulatory staff with the cooperation of the Commission's Reactor Development and Technology staff and national laboratories has developed a list of supplementary criteria for ASME Code-Constructed Nuclear Pressure Vessels. The regulatory staff has worked closely with the ACRS in the development of the criteria, and they reflect ACRS review and comment. Their purpose is to help assure that pressure vessels of licensed nuclear power reactors are built to the highest quality standards practicable. They reflect to a considerable degree current practice for pressure vessels of water-cooled power reactors, but are considered to be generally applicable to pressure vessels of other power reactors as well.

As a matter of convenience, each of the criteria has been related to a particular paragraph or part of the ASME Nuclear Vessel Code (1965) and published addenda, since this Code is usually specified for pressure vessels of nuclear power reactors by applicants for AEC construction permits. However, some of the criteria have been derived from other industry standards and specifications, such as those of the American Society of Testing Materials and the Society for Nondestructive Testing. In addition, in some instances the criteria include matters which are beyond the scope of present industry codes. These matters are of significance to the safety objectives and long-term reliability required for pressure vessels of nuclear power reactors.

Over the past several months, industry code groups have made considerable progress in upgrading existing codes and standards applicable to nuclear components, and they are actively considering further steps in this direction. These criteria may be useful to these groups in their efforts. Pending further development, the supplementary criteria are expected to be useful to pressure vessel designers and manufacturers and to others in the nuclear industry as interim guidance concerning AEC regulatory requirements for nuclear pressure vessels.

GENERAL REQUIREMENTS

§1.10 Classification of Nuclear Vessels. Pressure vessels in the reactor coolant system of pressurized water reactor (PWR) and boiling water reactor (BWR) plants shall be classified as follows:

PWR Plants

Reactor Vessel	Class A
Steam Generators (shell and tube side)	Class A
Pressurizer	Class A
Pressurizer Relief Vessel (or Quench Tank)	*Class C'
Regenerative or Excess Letdown Heat Exchangers (Chemical and Volume Control System)	Class A
Letdown Coolers (High Pressure Injection and Purification System or Chemical and Volume Control System)	Class A
Drain Coolers (Chemical and Volume Control System)	*Class C'
Reactor Coolant Purification Demineralizers	*Class C'
Reactor Residual Heat Removal or Shutdown Cooling Exchangers	*Class C'
Radioactive Waste Disposal System Vessels (Subject to pressures greater than would prevail if vented to atmosphere)	*Class C'

BWR Plants

Reactor Vessel	Class A
Regenerative Heat Exchangers (Primary System)	Class A

*Class C' requires compliance with rules of Subsection C of the Code for Class C vessels and the supplemental requirements of Code paragraph N-2113. Class C' vessels may be optionally reclassified as Class A vessels.

Reactor Coolant Purification Demineralizers *Class C'

Shutdown - Containment Spray Heat Exchangers (Reactor Shutdown Cooling System) *Class C'

Radioactive Waste Control System Vessels (Subject to pressures greater than would prevail if vented to atmosphere) *Class C'

These requirements supplement Code paragraph N-130.

Explanation - The quality of Code-constructed pressure vessels is dependent upon the classification selected (i.e., Class A or Class C'). Selection of appropriate vessel classification requires consideration of the operating conditions to which the vessel will be exposed and the nature of the safety functions which it will be required to perform to protect the public health and safety.

Code rules which would assure appropriate and consistent classifications of all vessels in nuclear power plants have not been developed. This criterion classifies the principal nuclear vessels whose performance during their service lifetime is essential to the protection of public health and safety.

§1.11 Conditions for Design. For pressure vessels classified as Class A or Class C' under §1.10, the Design Specification shall set forth the conditions for vessel design associated with:

(a) Normal Operating Conditions - Conditions to which the vessel will be exposed during normal operation of the facility (e.g., for a reactor vessel, the conditions include criticality, warmup, cooldown, operation from partial power level up to and including the anticipated maximum overpower level, and the expected transients in changing from one normal condition to another).

*Class C' requires compliance with rules of Subsection C of the Code for Class C vessels and the supplemental requirements of Code paragraph N-2113. Class C' vessels may be optionally reclassified as Class A vessels.

(b) Abnormal Conditions - Conditions not expected during normal service but to which the vessel will be exposed as a result of equipment failures, operating personnel errors, system load disturbance, or postulated malfunctions of components (e.g., for a reactor vessel, these conditions include (1) reactivity excursions due to inadvertent control rod withdrawal, program error, or component malfunction, etc., (2) interruption or partial loss of core coolant flow, (3) depressurization by active elements (e.g., relief valves), (4) malfunctions or failures in the steam or power conversion system, (5) reactor-turbine load mismatch or turbine trip).

(c) Fault Conditions - Conditions associated with extremely low probability events but which the vessel must be designed to withstand without loss of integrity because of their potentially serious consequences, (e.g., for reactor vessels, the fault conditions include those postulated accidents which may transmit undue static or dynamic loadings and blowdown forces onto the vessel, such as a major rupture of a reactor coolant system component (or in associated systems) or ejection or drop of maximum worth control rod).

(d) Environmental Conditions - Natural or service environmental conditions which when considered in conjunction with (a), (b), and (c) above may influence vessel design (e.g., for reactor vessels, the environmental conditions include those associated with service environments and natural phenomena such as (1) instability of vessel materials which may develop during service such as strain-aging, temper embrittlement, hydrogen embrittlement, etc., (2) anticipated changes in mechanical properties of the vessel material from irradiation exposures during service lifetime, (3) mechanical and hydraulic shock or vibratory forces transmitted to the vessel from postulated component malfunctions and faults originating in system components or from coolant flow-induced effects, and (4) seismic ground accelerations which have an expectancy of occurring in the vicinity of or at the plant site).

(e) Cyclic Conditions - For each pressure, thermal, and mechanical transient which a vessel may be subjected to under the conditions of (a), (b), (c), and (d), the cyclic conditions and their expected number of occurrences over the design service life of the vessel shall be specified in the Design Specification. The transients considered shall include both preoperational and such other hydrostatic or pressure tests which the vessel may be subjected to during its design life, whether imposed on the vessel alone or on the system of which it is a part. Transients associated with safety actions as imposed by the operations of engineered safeguard systems shall be included.

The number of cycles specified for each transient shall be conservatively estimated and shall be considered as the limit of occurrences permitted during the vessel's service life. The design cycles shall be specified in sufficient detail to enable the plant operator to identify and log the service cycles during plant operation over the vessel's service life.

These requirements supplement Code paragraph N-141.

Explanation - Because the safety functions associated with the reactor coolant systems must not only be reliably performed under normal operating conditions but also under abnormal situations, postulated design basis accidents, and environmental forces, the "conditions of design" for nuclear vessels must take into account all these conditions if safety requirements are to be met.

To meet this objective, Code-constructed nuclear vessels must be designed to withstand without impairment of their structural integrity the conditions identified in this criterion in addition to the conditions specified by Code rules.

§1.12 Certification of Stress Report.

(a) In addition to Code-required certification, the registered Professional Engineer(s) certifying the Design Specification shall review, or cause to be reviewed by engineers responsible to him, the Stress Report prepared by the vessel manufacturer and shall certify that the conditions of design specified in the Design Specification have been correctly interpreted and applied in the Stress Report. This certification shall be appended to the Stress Report with the Code-required certification.

(b) In addition to certification of the Stress Report by the vessel manufacturer or its design agent, the vessel owner or its agent shall provide, or cause to be provided, an independent review of the Stress Report by a registered Professional Engineer(s) competent in the field of pressure vessel stress analyses who shall certify with respect to:

(1) The applicability of the analytical methods employed as related to the conditions of design and design configurations.

(2) The acceptability of the assumptions, loading combinations, boundary conditions, and mechanical properties of materials as applied in the analyses for the service conditions specified in the Design Specification.

(3) The extent of design agreement with similar analyses for components of comparable vessels with similar service conditions.

The certification shall be appended to the Stress Report with the Code-required certification.

These requirements supplement Code paragraphs N-141 and N-142.

Explanation - To provide assurance that the design of a nuclear vessel satisfies the safety requirements applied to the nuclear power plant, the Code-required Stress Report which documents the vessel stress analyses must reliably reflect the correct application of the "conditions of design" as specified in the vessel's Design Specifications. It is incumbent upon the engineers responsible for preparation of the Design Specifications to verify the application and interpretation of the conditions of design as employed in the Stress Report.

In addition, to maintain the high quality standard in nuclear vessel design, the Stress Report must be subjected to an independent review if its adequacy in meeting the vessel's safety requirements is to be assured.

§1.13 Conditions with Unspecified Design Rules. Stress analyses shall be made for conditions for which design rules are not specified in the Code (e.g., mechanical shock, vibration effects, and dynamic loads). These stress analyses, including the design criteria, shall be identified in the Stress Report together with the bases upon which the structural capability of the vessel to withstand these vessel loadings are established.

This requirement supplements Code paragraph N-142.

Explanation - Nuclear vessels are subject to unusual conditions which impose mechanical shock, vibrations, or dynamic loads for which Code design rules are not available. Since the long-term reliability and safety of nuclear vessels may be influenced by the loadings imposed under these conditions, the vessel designer must establish conservative design criteria and perform appropriate stress analyses.

§1.14 <u>Vessel Owner's Responsibility for Inspection</u>. For Class A vessels, in addition to the Code-required inspections it shall be the responsibility of the vessel owner or its agent to employ and maintain one or more qualified owner's representatives at the vessel manufacturer's plant on a continuing basis during the course of vessel manufacture, and in the field during installation, to make or witness those inspections and review and verify those reports which are essential to assure that vessel construction is in accord with the requirements of the Design Specifications, material specification, approved fabrication drawings, and inspection and testing procedures, as implemented by the vessel manufacturer's quality assurance program.

The vessel manufacturer shall permit access for surveillance by the vessel owner's authorized representatives to any place where vessel design, material manufacture and storage, vessel fabrication, assembly, inspection, and testing are performed.

The vessel owner or its agent shall review the quality assurance program of the vessel manufacturer, and shall, if necessary, impose specific additional requirements to assure itself that adequate quality in manufacture will be attained.

The inspections shall not be considered complete until the vessel is fully installed (or erected), including all internals and piping connections at the installation site, and subjected to the final preoperational hydrostatic test of the reactor coolant system of which it is a part.

Such inspections shall not relieve the vessel manufacturer of the responsibility for the structural integrity of the vessel to the extent prescribed by the Code and with which it must certify compliance.

These requirements supplement Code paragraph N-143.

> Explanation - The most important contribution to assurance of attainment of the quality standard of nuclear vessels is the establishment and continued enforcement of all rules and requirements of design, materials, fabrication, inspection, and testing prescribed to achieve the intended final quality of the finished vessel.
>
> Since the ultimate responsibility for the safe and reliable operation of nuclear power plants rests with the vessel owner, it is incumbent upon the vessel owner to assure himself that all procedures and practices in the course of vessel manufacture are being competently performed without deviations from acceptance standards.

§1.15 Manufacturer's Responsibility for Quality Assurance. For Class A vessels, the manufacturer shall have a quality assurance program, including an adequate administrative and technical support organization.

The quality assurance program shall embrace all phases of manufacturing to assure a high level of quality throughout all areas of performance: design, development, materials, fabrication, processing, assembly, inspection, test, equipment maintenance, and handling for shipment. The program shall include the Code requirements specified in Appendix IX - Section IX - 200 - Quality Control System Requirements.

The vessel manufacturer shall make readily available to the vessel owner or its agent the written procedures and records of its quality assurance program as evidence of conformance with specified quality standards.

This requirement supplements Code paragraph N-144.

Explanation - Quality assurance in the manufacture of nuclear vessels must be extended to all facets of performance if the intended high level of quality is to be achieved in the finished vessel.

Such assurance must, of necessity, include the conduct by the vessel manufacturer of a program which verifies the appropriateness of the vessel design, the adequacy and approval of the stress analyses, the properties and soundness of the vessel's materials, the procedures for each fabrication operation, the monitoring of each fabrication step, the resolution of manufacturing deviations, and the competency of the performance of all inspection and testing practices.

Evidence of such conformance is a prerequisite in determining the acceptability of the fabricated vessel and provides the required measure of assurance of the quality and safety of nuclear vessels.

§1.16 Vessel Fabrication Report. The vessel manufacturer shall prepare a Vessel Fabrication Report within six months of completion of fabrication of a Class A vessel. The report shall be certified by the vessel manufacturer with respect to the accuracy of the contained information after an audit performed by the vessel owner's representatives present during the course of vessel manufacture. The report shall be made available to the vessel owner who shall assume the responsibility of maintaining the report on file for the period of the vessel's service life. The Vessel Fabrication Report shall include, at least, the following:

(a) Mill test reports of all materials within the vessel's pressure boundary, including the heat treatment data and the Charpy impact test results of the material test coupons.

(b) The written weld procedure qualifications, including the results of mechanical properties tests, Charpy impact tests, and metallurgical examinations performed on test specimens and weld materials.

(c) The written nondestructive examination procedures, including any additional requirements and acceptance criteria beyond those specified in the Code.

(d) Material and weld joint repairs and postweld heat treatments performed in the course of manufacture accompanied by identification and location of such repairs on the vessel drawings.

(e) All manufacturing deviations, which occurred during any phase of fabrication, and the corrective actions or dispositions taken, as approved by the vessel owner or its agent.

(f) A detail record of findings from all final nondestructive examinations (radiographic, magnetic particle, liquid penetrant, and ultrasonics) performed on the vessel or vessel components. For vessel components not accessible during the final vessel examination, the record of the last examination performed at an earlier stage shall be included. The records shall be adequate to serve as a reference examination for comparison with future examinations as may be required during the service life of the vessel.

(g) Vessel flange bolt tightening procedures and preloads and bolt elongation measurements taken during assembly at the manufacturer's plant, which are required for bolting operations during the service life of the vessel.

This requirement supplements Code paragraph N-144.

Explanation - In recognition of the long-term reliability and safety expected of nuclear vessels, examinations of these vessels at periodic intervals during its service life may be required. To enable a meaningful assessment of the structural integrity of the vessel following such examinations, a fabrication history of the vessel is essential to evaluate any unexpected structural deterioration or damage sustained in service.

The Vessel Fabrication Report serves as a reference upon which the adequacy of the vessel for continued service may be assessed by comparison with the records of the examinations performed during the vessel manufacture.

§1.17 Boundary Between Vessel and Piping. The control rod housings of a reactor vessel shall be considered as extensions of the vessel's pressure-retaining boundary and the rules of Subsection A of the Code shall apply to that portion of control rod housings which are exposed or may be exposed to the reactor coolant pressure.

This requirement supplements Code paragraph N-150.

Explanation - Control rod systems of reactor vessels constitute a group of appurtenances directly connected to the reactor vessel. The safety and reliability of the control rod housing in service are of paramount importance to the safe operation of the reactor vessel. It is, therefore, essential that the standard of quality of the reactor vessel be extended to the control rod housings.

MATERIALS

§1.20 Vessel Material Property Improvement. For Class A vessels, the material specifications of ferritic materials of any product form (wrought or cast) to be used in the pressure-retaining boundary shall require aluminum killing and vacuum degassing treatment in manufacture, or other treatments producing comparable material property improvement.

For reactor vessel ferritic materials which are intended to directly surround the reactor core where the neutron fluence is above 10^{17} nvt (E_n of 1 Mev or above), the material specification shall limit the phosphorous content to 0.012 percent maximum and the sulfur content to 0.015 percent maximum for both ladle and check analysis.

This requirement supplements Code paragraph N-310.

> Explanation - The reliance placed upon the materials of construction of nuclear vessels to retain their physical and mechanical properties over long intervals of service without jeopardy to the vessel's structural integrity demands the selection of high quality materials in vessel manufacture.
>
> To attain the level of quality expected of nuclear vessel, it is essential to require manufacturing practices which produce cleaner steels with improved metal fatigue properties and less susceptibility to the detrimental effects of strain-aging and material embrittlement under service conditions. Improvements in material quality are achieved by the application of vacuum degassing processes employed during material manufacture as well as by more rigid controls of the chemical composition than applied to materials for nonnuclear applications.

§1.21 Material Test Coupons. Material test specimens shall be taken from the end of each mill rolled plate which represents the top end of the ingot.

This requirement supplements Code paragraph N-313.4(a).

Explanation - In large ingots intended for components of nuclear vessels, material properties vary substantially. To assure the selection of vessel materials throughout the vessel plates which meet the minimum material specification properties, it is essential to remove specimens from those areas of steel plates representative of the poorest quality of the ingot for the purpose of verifying their physical and mechanical properties.

§1.22 Nondestructive Examination of Reactor Vessel Plates.

(1) Ultrasonic Examination - In addition to the inspection employing the straight beam technique, all plates for reactor vessels shall be ultrasonically inspected over 100% of the plate surfaces using a 45° angle beam or shear wave technique, both longitudinally and transversely to the major plate rolling direction.

The examinations shall be performed on the shell courses and head segments of the vessel after final forming operations and any heat treatment employed directly upon completion of forming but prior to welding of the shell courses or head segments. Vessel plates subject to an accelerated cooling phase of the heat treatment to enhance properties shall be ultrasonically examined after accelerated cooling.

(2) Test Surface for Angle Beam Test - The test surface shall contain a machined calibration notch with a 60° included angle whose depth is equal to 2% of the plate thickness and whose length is between 1/2" and 3/4". The test surface shall not be part of the vessel pressure boundary nor closer than 2" to any edge.

(3) Acceptance Standard for Angle Beam Test - Any ultrasonic indication equal to or exceeding that obtained from the calibration notch shall be

cause for rejection or repair in accordance with requirements of the Code and criterion §1.23.

These requirements supplement Code paragraph N-321.1.

Explanation - The manufacturing difficulties in maintaining steel quality generally increase with the plate thicknesses. The use of heavy steel plate thicknesses in nuclear reactor vessels introduces the need to verify the quality of these plates prior to vessel fabrication. The technique of nondestructive examination of the materials provides the means for locating significant and unacceptable manufacturing defects.

Because of the importance of the safety functions associated with the reactor vessels in nuclear power plants, it is essential to adopt examination techniques for vessel materials which will assure elimination of defective materials.

Defects can be introduced in vessel steel plates both during their manufacture and during vessel fabrication processes such as forming and heat treatment. Examinations which follow these processes are more likely to reveal unacceptable flaws.

§1.23 Nondestructive Examination and Repairs of Materials. Areas repaired by welding in materials intended for Class A vessels shall be radiographically examined following the postweld heat treatment. Such welds shall meet the acceptance standard applied to vessel welds in accordance with Code rules and as supplemented by criterion §1.51.

The welding procedure and the welders or welding operators employed by the manufacturer of materials (in any product form) in making weld repairs shall be qualified in accordance with Section IX of the Code and meet the following applicable requirements of Section III of the Code with respect to:

1) N-320 - Nondestructive Examination and Repairs of Material;

2) N-520 - Welding Processes, Weld Qualifications and Records, and Precautions for Welding;

3) N-530 - Preheating and Postweld Heat Treatment;

4) N-620 - Inspection of Welding and Acceptance Standards; and

5) Appendix IX - Quality Control and Nondestructive Examination Methods.

The material manufacturer shall make available to the vessel owner or its agent, upon request, records of the written welding qualification procedures used in making repairs, repair procedures, extent of repairs, results of nondestructive examinations, and certification of compliance with the applicable rules of Section III of the Code.

These requirements supplement Code paragraphs N-321.2, N-322.4, N-323.5, and N-324.9.

> Explanation - In order not to degrade the high quality of the finished vessel, it is essential that nondestructive examination requirements and acceptance standards for material repairs be equal to those required for welding of the fabricated vessel.

§1.24 Examination of Reactor Vessel Bolts. Nondestructive examinations of the bolts for reactor vessel flange closure shall be performed on the finished component after completion of threading operation and heat treatments.

The liquid penetrant examination shall be in accord with Code paragraph N-627, except that a high-sensitivity post-emulsifiable fluorescent penetrant shall be used, and shall meet the acceptance standard of N-325.2.

The ultrasonic examination shall be performed in accordance with N-625.3 and meet the acceptance standards of N-325.3.

Where threads are subject to surface treatment or plating processes, the examination shall be performed both prior to and after surface treatment.

These requirements supplement Code paragraph N-325.

Explanation - Closure studs and bolts of reactor vessels constitute critical components whose structural integrity is relied upon during the entire service life of the vessel to the same extent as the materials that form the pressure-retaining boundary of the vessel.

Nondestructive examinations of these components upon the completion of all manufacturing operations and heat treatment are essential to reveal defects which could potentially contribute to loss of integrity or failure under service loading conditions.

§1.25 Ductile Brittle Transition Properties. For reactor vessels, the ductile brittle transition properties of ferritic materials shall conform with the following requirements:

(a) Impact-absorbed energy values of all carbon and low alloy steel intended for the main closure flanges and the shell and head materials connecting thereto, shall meet the impact test values specified in Code Table N-421 at a temperature no higher than 10 F.

(b) The properties of carbon and low alloy steel intended for the shell materials directly surrounding the reactor core shall satisfy the following requirements:

(1) A ductile to brittle transition (NDT) temperature no higher than 10 F as determined by drop weight tests conducted in accordance with Code paragraph N-331.1.

(2) The impact test value specified in the Code Table N-421 at a test temperature no higher than 10 F as determined by Charpy-V-notch tests conducted in accordance with Code paragraph N-331.3.

(c) Impact-absorbed energy value of carbon and low alloy steel pressure-retaining material not specified in (a) or (b) above, and of the material for the vessel support skirt shall meet the requirements specified in Code Table N-421 at a temperature no higher than 40 F.

Where Charpy-V-notch specimens are used for the impact tests of (a), (b), and (c) above, impact-absorbed energy values shall be determined from specimens taken in a plane parallel to the material surface with the long axis of the specimen parallel to the direction of the major rolling or forging operation, and at a location with respect to material thickness and heat treated edge as specified in Code paragraph N-313.4.

In addition to the test specimens required for (a), (b), and (c) above, at least 9 additional Charpy-V-notch specimens from each heat of the materials shall be used to determine the temperature region of transition from ductile to brittle fracture and the energy absorbed in the region of 100% shear fracture. The upper shelf absorbed energy shall, as a minimum, meet the following requirements:

(d) For materials directly surrounding the reactor core, including welds and weld heat-affected zones, the upper shelf absorbed energy test value of any longitudinal specimen of carbon and low alloy steels shall be no less than 60 ft.-lbs. at a temperature no higher than 160 F.

(e) For materials of formed heads, including welds and weld heat-affected zones, the upper shelf absorbed energy test value of any transverse specimen of carbon and low alloy steels shall be no less than 40 ft.-lbs. at a temperature no higher than 160 F.

(f) For bolting materials, the upper shelf absorbed energy test value of any longitudinal specimen shall be no less than 40 ft.-lbs. at a temperature no higher than 160 F.

All impact-absorbed energy values of (d), (e), and (f) shall be determined from Charpy-V-notch specimens taken in a plane parallel to the material surface with the long axis of the specimen parallel to the direction of the maximum principal stress which the material will be subjected to in service, and at a location with respect to thickness and heat treated edge as specified in Code paragraph N-313.4. For plates used in formed heads, the direction of the maximum principal stress shall be considered to coincide with the direction of the minor rolling operation.

These requirements supplement Code paragraph N-331.

Explanation - The major irradiation-induced changes in the mechanical properties which occur in ferritic steels require that the reactor vessel materials possess properties (ductile-brittle transition) which provide a safe margin for operation from the range of conditions where the potential for a brittle mode of vessel failure exists.

To provide this margin, it is essential to select materials whose initial ductile-brittle transition characteristics are sufficiently conservative to accommodate the expected irradiation-induced embrittlement in service without imposing unacceptable operating limitations and without jeopardizing the safety of the reactor vessel during service.

§1.26 Exclusion of Repairs in Bolting Materials. Bolting materials with defects requiring repairs by welding shall be unacceptable for vessel flange closure studs, bolts, and nuts.

This requirement supplements Code paragraph N-322.4.

Explanation - Bolting materials for reactor vessel closure flanges are subject to cyclic loading conditions in service which may adversely influence the metal fatigue life of the material. The presence of defects or weld repairs in the materials of these critical vessel components reduces their long-term reliability. To obtain the high level of bolting integrity for safe operation of the reactor vessel, it is essential to select bolting materials free of defects or weld repaired areas.

DESIGN

§1.30 Fracture Mechanics Analyses. For reactor vessels which may be exposed to a neutron fluence in excess of 10^{17} nvt (E_n of 1 Mev and above), an analysis shall be performed to estimate the margin between the crack size as a result of growth under design cyclic loads and the critical crack size for brittle fracture in the welds of the vessel shell material which directly surrounds the reactor core region. The analysis shall be based on the growth rate of anticipated flaws under design cyclic loads and on material properties at a temperature 60 F above the nil-ductility transition temperature. The critical crack size shall be calculated for each period during which the material properties may significantly change to affect the results.

The analysis shall demonstrate that the estimated fatigue crack size at any time in the vessel service life will be significantly less than the critical crack size for brittle failure.

This requirement supplements Code paragraph N-415.

> Explanation - Nuclear vessels are subject to transient loads of cyclic character which may cause flaws in critical weld zones to grow.
>
> In evaluating the severity of fatigue crack growth in zones of the reactor vessel where irradiation tends to embrittle the material, it is in the interest of safety to estimate the margin between the ultimate size of the fatigue crack under repeated load variations and the critical size for fracture. Fracture mechanics provides the principles upon which this brittle fracture potential may be assessed.

§1.31 <u>Design for Cyclic Loading</u>. The Code design fatigue curve of Figure 415(a) for reactor vessel components subject to a neutron fluence in excess of 10^{17} nvt (E_n of 1 Mev and above) in service shall be modified by reducing the allowable amplitude of alternating stress intensity, Sa, by 25 percent.

This requirement supplements Code paragraph N-415.2(c).

> <u>Explanation</u> - Neutron irradiation of ferritic steels in reactor vessels causes changes in ductility of the material during service which introduces uncertainties, as yet undefined, with respect to the low cycle fatigue resistance of the steels.
>
> Nuclear vessel may contain cracks or flaws of a size below the threshold of detection by the nondestructive examination techniques employed during fabrication. It is essential to provide an increased safety margin for materials in an irradiation environment beyond the margin required for nonirradiated materials.

§1.32 <u>Bolting Design Requirements</u>. The design of bolted connections for Class A vessels shall take into account the provisions necessary to facilitate periodic examinations of the bolting or studs during service lifetime, and bolting or stud replacement if required.

Thread roots shall be appropriately radiused and machined or rolled to a fine finish to reduce stress concentrations.

Closure studs two inches and larger in diameter shall be designed to accommodate any anticipated rotation of flange faces during initial bolt tightening operations in order to limit stud bending within the Code allowable design stress intensities (e.g., spherical washers).

Bolting design shall provide for the use of bolt tighteners which axially elongate the bolts under controlled preload conditions and enable precise measurement of both the applied preload and the actual bolt elongation. The bolt tightening procedures, preloads, and bolt elongation measurements shall be recorded in sufficient detail to enable all subsequent bolting operations during the service life of the vessel to be performed within the prescribed limits. Such information shall be contained in the appropriate section of the Vessel Fabrication Report (Criterion §1.17).

These requirements supplement Code paragraph N-416.

> Explanation - The bolts for flanged closures of nuclear vessels are components which, by design, have areas of high stress concentrations. The load carrying capability of the bolting is vital to the safety of the vessel.
>
> Unless design provisions enable both in-service inspection and replacement of bolts or studs, the development of cracks in bolting under cyclic loading may jeopardize the structural integrity and the continued safe operation of nuclear vessels.

§1.33 Earthquake Loading. Where earthquake loadings are specified in the Design Specifications, the determination of the seismic-induced stresses shall be based upon the application of acceptable methods of dynamic analysis for the calculation of the structural response of the vessel to earthquake motions. The analysis shall take into account the response spectra of the ground motions, the degree of structural damping, and the amplification of ground motions as dictated by specific site conditions.

In determining the maximum stresses, the effects of vertical components of seismic motion shall be combined directly and linearly with the effect of horizontal components of earthquake motion, and both vertical and horizontal components shall be combined directly and linearly with other loadings specified under the criterion of §1.34.

The cyclic loading associated with design seismic-induced vibrations shall be included in the fatigue analysis.

Consideration shall be given to out of phase displacements of the vessel supports, or components of vessels (e.g., control rod assemblies on reactor vessels, connected piping, etc.) resulting from differences in seismic-induced motions of vessels, components, and appurtenances connected thereto, and to the possibility of tilting or rotation of structural foundations upon which the reactor vessel rests.

This requirement supplements Code paragraph N-447.

> Explanation - A principal safety requirement for a nuclear power plant is the assurance of the capability for a safe and secure shutdown of the facility in the event of an earthquake occurring at the plant site. Such a capability must be provided for by designing nuclear power plant components (i.e., vessels) to resist the design basis earthquake without impairment of their structural integrity.
>
> Because of the uncertainties associated with the effects of earthquake loadings on nuclear power plant components, it is imperative that safe shutdown be reliably achieved in order to render the plant secure for the protection of public health and safety. This shutdown capability is also essential to reverify the functional operability of the protective systems and engineered safeguards for the reactor coolant system prior to resumption of plant operation.

§1.34 Design Conditions - Combinations of Loadings. Class A vessels and their supports shall be designed on the basis of the loadings imposed by (a) normal operating conditions, (b) abnormal conditions, (c) fault conditions, and (d) environmental conditions. These conditions are identified under criterion §1.11.

The vessel and its supports shall be designed to accommodate the most severe loading combinations which may act simultaneously. The combinations of loading shall include but not necessarily be limited to those imposed by the following combination of conditions:

(1) Normal operating conditions plus any system transients in changing from one normal condition to another.

(2) Normal operating conditions plus any system transient imposed by the development of abnormal conditions (emergency or upset system condition).

(3) Normal operating conditions plus any system transient resulting from the occurrence of postulated system fault conditions (system component failure).

(4) Normal operating conditions plus environmental forces (earthquake where specified).

(5) In addition to the vessel loading combinations specified in (1), (2), (3), and (4), a reactor vessel, its external supports, and the internal reactor core and structure supports shall be designed to accommodate the most severe coincident loadings associated with:

a. The dynamic loads imposed on the vessel by the design basis earthquake at a time when the reactor is operating at full-rated power, and

b. The system transient loads transmitted to the vessel upon a postulated severance of any connected piping or upon a failure of a reactor coolant system component assumed to occur in consequence of the design basis earthquake of (a), and

c. The transient forces transmitted to the vessel via the supports of the reactor core and other vessel internal structures as a result of the sudden depressurization caused by the system transient of (b).

For loading combination (1) and (2), the design stress intensities shall be in accord with the values specified in the Code.

For loading combinations (3) and (4), the design stress intensities for general primary stress shall not exceed 90 percent of specified minimum yield strength of the vessel steel at maximum operating temperature (Sy value from Code Table N-424).

For loading combination (5) the principles of limit analysis may be applied, in which case, the combined loadings shall be limited to 90 percent of the lower bound limit for yield collapse load (based on the maximum shear stress failure criterion and the Sy value specified in Code Table N-424 corresponding to the maximum vessel operating temperature). If the yield collapse load is not determinate, tests may be performed to define its value for the specified loading combination. In such tests, the collapse load shall be taken as that combination of loading when the measured strain is two times the strain value at the point of the initial deviation from the elastic stress-strain linearity.

These requirements supplement Code paragraph N-447.

> Explanation - The safety functions assigned to nuclear vessels are essential not only under normal operating conditions, but also under the combination of conditions or a simultaneity of forces which may prevail in the event of anticipated system malfunctions, postulated failures in reactor coolant system components, and the multiple effects of earthquake shocks and system ruptures.

> Nuclear vessels constitute those components of nuclear power plant which contain the major portion of the system energy with the greatest potential for destructive forces upon failure. To protect the public health and safety, they must be designed to accommodate the most severe combination of loadings without failure.

§1.35 Computer Programs. Analytical design techniques may employ computer programs provided their applicability is appropriately established in the Stress Report, and the results are validated by comparison with proven analytical methods of stress analyses, other verified computer programs, or experimental procedures. Experimental stress analysis in compliance with Code Article 1-10 is an acceptable method in validating the use of new computer programs.

The information presented in the Stress Report with respect to computer programs employed in the vessel stress analyses shall be sufficient to enable independent verification of the input data, the analytical model adopted, the assumptions, and boundary conditions as they relate to the conditions for vessel design.

This requirement supplements Code paragraph N-432(b).

> Explanation - Within the present state of the pressure vessel design technology, many analytical solutions to design problems have been computerized to reduce the work of stress analyses. However, the mathematical complexity associated with the commonly applied linear theory of elasticity has proven formidable in providing exact solutions to all pressure vessel design problems of interest. The analytical solutions derived from approximate theories with simplified assumptions based on gross structural behavior and empiricism can only be verified by comparison with proven analyses or experimental investigations.

The validity of the analytical solutions cannot readily be verified by measurements of stresses and strain on the completed nuclear vessels. Neither can such measurements be taken in a practical manner after the vessel is placed in service. In consequence, limited data of vessel structural response are available. Design predictions must therefore depend heavily upon the adequacy and accuracy of the analytical methods employed by the vessel stress analyst.

§1.36 Environmental Effects

(a) Irradiation-induced Effects - For reactor vessels of ferritic materials, where the expected neutron fluence over the specified life exceeds 1×10^{17} nvt (E_n of 1 Mev and above), the design shall make provisions for the placement of material surveillance specimens in the vessel for the purpose of monitoring and evaluating, at periodic intervals, the radiation-induced material property changes and to establish as required, limitations on operating conditions. The design shall accommodate sufficient material surveillance specimens conforming with ASTM Standard E 185-66T, and enable surveillance tests to be performed at intervals of 1/4, 1/2, and 3/4 of the vessel's service lifetime.

(b) Time-dependent Effects - For reactor vessels, the design shall take into account the time-dependent effects of deteriorative factors (i.e., corrosion fatigue, creep instability, strain-aging, etc.) under operating conditions for the specified vessel life. Whenever anticipated changes of the initial mechanical or physical properties of the materials toward the end of vessel life may adversely influence their serviceability, the design shall accommodate material surveillance specimens to monitor the progress of these deteriorative factors.

This requirement supplements Code paragraph N-446.

> Explanation - Based on currently available data and experience, accurate predictions of the environmental effects such as irradiations and service conditions on nuclear reactor vessel materials are either uncertain or subject to significant error. To assure that the mechanical properties remain within the acceptable range for safe operation of the nuclear vessel, it is necessary to employ means of monitoring changes which may occur in service.

§1.37 Design for Inspectability. The design of the reactor vessel shall provide accessibility for visual inspection at appropriate intervals during its service lifetime of all critical areas and the interior surfaces of the vessel, including the bottom head. Critical areas include structural discontinuities and the principal weld joints of the vessel. The attachments to the inside of the reactor vessel shall be designed to enable removal of all internal components necessary to permit visual inspection of the interior surfaces of the vessel aided by remotely operated optical equipment where necessary.

The provisions of accessibility for inspection shall further enable examination of essentially 100 percent of the volume of the reactor vessel material, either from the inside or outside surfaces of the vessel or a combination thereof, by ultrasonic or other methods. The extent of the vessel subject to examination shall include all welds within the vessel boundary up to and including the welds of transition sections between the vessel nozzles and the connected piping, and the welds of the reactor control rod housings to the vessel head.

The design of Class A vessels, other than the reactor vessel, shall provide accessibility for inspection and examination of all critical areas either from the inside or outside surfaces of the vessel or a combination thereof.

These requirements supplement Code paragraph N-440.

Explanation - In recognition of the critical safety functions associated with the nuclear reactor vessel, the preservation of its structural integrity throughout its operating history is of primary importance to the safety of nuclear power plants. To demonstrate that continued operation of the reactor vessel at any time does not incur a risk of a rupture of the vessel, a program of periodic examination and inspection is necessary.

Because of the attendant difficulties associated with the inspection of reactor vessels in the presence of a radioactive environment, implementation of postoperational inspections requires consideration on the part of both the plant and vessel designers in developing designs which enable and facilitate these inspections.

§1.38 Attachments to Reactor Vessels

(a) For reactor vessels of ferritic materials, vessel nozzles shall not be located in any shell sections which directly surround the reactor core region and which is calculated to receive integrated neutron doses in excess of 1×10^{17} nvt (E_n of 1 Mev and above).

(b) Partial penetration welds, shown in Code Figure N-462.4(d) as applied to control rod housings of reactor vessels shall be limited to the inside of the vessel.

(c) Supports and supporting members attached to the reactor vessel wall by welding shall be located in areas other than the shell sections which directly surround the reactor core.

These requirements supplement Code paragraphs N-457(a), N-457(c), and N-473, respectively.

> <u>Explanation</u> - The attachment of appurtenances to nuclear vessels superimposes structural discontinuities which significantly alter the normal stress patterns in the walls of the vessel under load or which introduce undesirable stress intensifications.
>
> These areas of stress intensifications at the attachments introduce conditions susceptible to the crack development in service. In turn, these cracks may initiate either ductile or brittle fractures through the pressure-retaining wall of the vessel.
>
> Avoidance of attachments in the zones of the reactor vessel where the mechanical properties change during service is essential to eliminate these adverse stress conditions.

§1.39 <u>Reactor Vessel Core Support</u>.

(a) Attachments to the inside of a reactor vessel for support of reactor fuel core structure shall be designed to withstand the loadings under normal operating cycles, abnormal conditions, postulated fault conditions, and combination thereof as stated in criterion §1.34.

The attachments shall be designed to sustain the most severe of the loading combinations within the design stress intensities specified by the Code and within deflection limits which allow unimpaired control rod motion under such loadings as well as preclude mechanical damage to fuel assemblies.

Load-carrying attachments welded to the inside of the reactor vessel shall be designed to provide full penetration welds which enable either radiographic examination for their entire length or examination by means of ultrasonic techniques, magnetic particle, or dye penetrant methods specified in the Code. If the examination is performed by magnetic particle or dye penetrant methods, each weld layer shall be progressively examined.

(b) Where the internal supports of the reactor core structure are welded directly to the weld overlay cladding of the reactor vessel, the area of attachment shall be examined 100 percent by the ultrasonic technique prior to welding of the structural support. The ultrasonic examination shall be performed in accord with Code paragraph N-625.2, weld repairs in accord with Code paragraph N-625.4 and the acceptance standards of Code paragraph N-625.3 shall be met.

Requirements (a) and (b) supplement Code paragraphs N-474 and N-518.5, respectively.

> Explanation - The design of the structural attachments to a reactor vessel which provide the principal support for the reactor's nuclear fuel core must provide a reliable means of holding the core in position under any condition of loading, either anticipated or postulated which the reactor vessel may sustain.
>
> The loss of the reactor core supports could allow a disarrangement of the nuclear fuel assemblies, or the disengagement of the reactor core structure from the system of the control rods with a concomitant uncontrollable reactivity change. The consequences would carry the risk of overpressurization of the reactor vessel and crack development. To protect the public health and safety, this requires a design and fabrication of reactor core supports which provide the highest integrity in service.

FABRICATION

§1.40 Chemical Analysis of Weld Wire. A chemical analysis of solid and stranded wire filler metal shall be performed on each section of wire which may be spliced together in one coil. The wire coil shall be obtained from the manufacturer with not more than one splice to enable sampling for chemical analysis of the accessible ends as a practical means to verify that the entire wire coil meets specifications.

This requirement supplements Code paragraph N-511.5.

Explanation - Within the limits of the present day pressure vessel technology, the structural integrity achieved in the fabrication of the vessels depends substantially upon the welding processes employed to join and assemble the vessel components, and the properties of the welded joints.

Although many tests are conducted on weld materials prior to their welding application to verify their compliance with specifications, it is not practical to reverify the properties of completed welds on nuclear vessels. Because misapplication or the improper use of weld materials cannot be discovered, it is essential that all weld materials be tested in a manner which will preclude weld materials in nuclear vessels not in conformance with specifications.

§1.41 Cutting Plates and Other Products. Plates and other products cut to shape by thermal cutting in preparation for welding shall have the edge surfaces cleaned by mechanical means (machining, shearing, chipping, or grinding) before examination in accord with the requirements of Code paragraph N-513.2 and prior to welding.

At least 1/32 inch of metal shall be removed from all surfaces thermally cut by air-arc, inert gas-arc, or carbon-arc. Not less than 1/8 inch shall be removed where oxygen-arc or oxygen cutting (including both powder and flux cutting) is used.

The weld preparation and adjacent base metal surfaces for a minimum of one inch on each side of the weld preparation shall be smooth, clean, and free of any foreign matter. The weld preparation shall be protected from contamination until welding is started and fully completed.

This requirement supplements Code paragraph N-519.

> Explanation - Despite the practice of nondestructive examinations of materials for the purpose of detecting flaws prior to their acceptance, experiences have demonstrated that small flaws may escape detection, particularly at the edges of vessel materials which are to be joined by welding.
>
> The existence of any flaw in the welds of a nuclear vessel where unavoidable residual weld stresses and operating stresses coexist may pose a threat to the continued safety of the vessel in service. Adherence to practices which enable the elimination of unsound materials in the metal regions subject to welding is recognized as a prerequisite for sound welds.

§1.42 Welding Qualification Procedure Requirements.

(a) Test Material Requirements - In lieu of test material thickness requirements of ASME Boiler and Pressure Vessel Code Section IX-(1965), Tables Q-13.1 and Q-13.2, the requirements of the weld procedure qualification test of Code paragraph N-541.2, and of the impact tests of Code paragraph N-541.3(e) as applied to Class A vessels shall be met by welding test material obtained from one or more heats of the vessel, and whose thickness is equal to the major thickness of any joint in the vessel.

Alternatively, if the test material thickness requirements as specified in ASME Boiler and Pressure Vessel Code Section IX-(1965), Tables

Q-13.1 and Q-13.2, are used for the requirements of the weld procedure qualification test of Code paragraph N-541.2, only the requirements of impact test of Code paragraph N-541.3(e) need be met by welding test material obtained from one or more heats of the vessel, and whose thickness is equal to the major thickness of any joint in the vessel.

(b) Base Material - Weld procedure qualifications for reactor vessels of ferritic materials shall require the use of base material test plates meeting the impact test values in Code Table N-421 at a temperature no higher than 10 F. The procedure qualification test plate for weldments which are to be subject to an austenitizing heat treatment shall be at least 3t x 3t in size with the weld test coupons taken at least t from any edge of the plate.

(c) Metallurgical Examination - The weld procedure qualification report shall be accompanied by a report of a metallurgical examination, including photographs of the bend test specimens, of the tensile test coupons, and a photomacrograph of the etched cross section of the welds in the area where the tensile coupons have been removed.

The photomacrograph shall show weld bead sequence and groove design substantially similar to that specified in the procedure qualification. The photomacrograph specimens shall be subjected to a magnetic particle examination in accord with Code paragraph N-626, a liquid penetrant examination in accord with Code paragraph N-627 and meet the acceptance standards of Code paragraph N-320.

(d) Impact Test of Procedure Qualification Welding - For welding procedure qualifications intended for the welding of reactor vessel components, impact testing of the procedure qualification weld deposit and heat-affected zone shall meet the impact test values assigned the base material in Code Table N-421 at a temperature no higher than 10 F.

Requirements (a) and (d) supplement Code paragraphs N-541.1 and N-541.3(e), respectively. Requirements (b) and (c) supplement Code paragraphs N-541.2(a) and N-541.2(d), respectively.

> Explanation - Prior to the performance of production welding on nuclear vessels, the practice is to qualify the weld procedure in accord with prescribed rules and tests to ensure that the mechanical and metallurgical properties of vessel weld joints will be comparable to those of the vessel materials.
>
> The safety of nuclear vessels is directly and adversely influenced by any disparity between the properties of weld joints and the vessel material. Normal variations in the chemistry, mechanical and physical properties are experienced in both the vessel materials and weld metal. It is imperative to verify the quality of welds for each vessel as may be produced by combinations of different lots of the actual vessel materials and weld metals used if a consistent quality is to be assured in manufacture.

1.43 Precautions for Welding. All low alloy, low hydrogen electrodes, and fluxes used in welding shall be stored in a dry place.

The coated electrodes shall be baked for one hour in an oven at 800 F before use. After the baking cycle, the electrodes shall be stored in holding ovens maintained at 300 F plus/minus 50 F. Welders shall be permitted to remove only that amount of electrodes that can be used during a two-hour period.

Welding fluxes shall be stored in holding ovens maintained at 300 F plus/minus 50 F. Fluxes transferred to welding machines from the holding oven shall be maintained continuously at a temperature sufficient to preclude moisture absorption or shall be returned to holding oven, if unused.

This requirement supplements Code paragraph N-523(b).

Explanation - The storage conditions for weld materials are recognized as contributory to the resulting quality of weld joint in nuclear vessels. The importance of adopting acceptable practices as part of the quality control program for weld materials is emphasized by the experiences of poor quality of welds resulting from improperly maintained weld materials.

Since weld analyses are not practical after weld metal deposition in the nuclear vessel, precautionary controls of weld material must be relied upon to preclude degrading the quality of the completed vessel.

1.44 Welding Requirements.

(a) Weld Root Examination - In addition to the preparation requirements specified by the Code for the root of the second side of double-grooved joints welded from both sides, a magnetic particle inspection of ferritic materials in accord with Code paragraph N-626, or a liquid penetrant test for nonferritic materials in accord with Code paragraph N-627, shall be performed prior to the application of weld metal. The tests shall be followed by an appropriate cleaning procedure to remove all traces of materials used in conducting the tests.

(b) Preheating Requirements - The minimum preheating temperature to be maintained prior to any flame cutting operations and during the performance of category A, B, C, and D welds, including weld repairs, shall meet the following requirements:

P3 materials	- 1" or less in thickness	200 F
	over 1" in thickness	300 F

P4 materials -	3/4" or less in thickness	300 F
	over 3/4" in thickness	400 F
P5 materials -	3/4" or less in thickness	400 F
	over 3/4" in thickness	500 F

The preheating temperature shall extend at least 2t on both sides of the weld where t is the weld section thickness. The preheat shall be maintained until the vessel or component of the vessel is subjected to the postweld heat treatment. Any loss of preheat before completion of the weld shall require a weld surface magnetic particle inspection and a radiographic examination before resumption of welding.

(c) Requirements for Postweld Heat Treatment.

1) The postweld heat treatment temperatures existing throughout a vessel or component shall not differ from that employed in the weld procedure qualification test plate by more than minus 25 F or plus 50 F, taking into consideration the temperature tolerance measured in the test plate with respect to the minimum holding temperature specified in Code Table N-532.

2) A written heat treatment procedure shall be prepared detailing temperature, times, heating and cooling rates, thermocouple location, number of recordings and chart speed.

3) Temperatures shall be measured by the use of sufficient number of thermocouples attached to the vessel to assure that the temperature gradients do not exceed the limit of subparagraph (1) above. Such temperature measurements shall be autographically recorded and made available to the vessel owner or its agent upon request.

4) The temperature measuring equipment shall be calibrated at least

once each month. Records of calibration shall be made available to the vessel owner or its agent upon request.

Requirement (a) supplements Code paragraph N-527.1, requirement (b) supplements Code paragraph N-531, and requirements (c) supplement Code paragraphs N-532.3 (3) (6) (7) (8) respectively.

> Explanation - The design of the major strength welds in a nuclear vessel requires the application of weld metal from both the inside and outside of the vessel, principally because of the heavy wall thickness which must be joined.
>
> Experiences have demonstrated the need to examine by nondestructive techniques the weld deposit at the root of the joint to preclude weld defects from lack of fusion or penetration.
>
> Additionally, nuclear vessel materials require not only preheating to established temperatures to lessen their susceptibility to cracking or microfissuring as the weld joints are completed, but also closely controlled postweld heat treatment to enhance their mechanical and metallurgical properties after welding.
>
> To achieve the high quality expected of nuclear vessels, closely controlled weld preheating and postheating is a necessary practice to preclude weld defects detrimental to the vessel's safety.

INSPECTION

§1.50 Final Inspection and Examination. In addition to the nondestructive examinations required by Code paragraph N-618.2, the vessel shall be subjected, after completion of the hydrostatic test, to an ultrasonic examination of all accessible weld surfaces of the pressure boundary including the weld clad surfaces. The examination shall be performed to provide for 100 percent volumetric inspection of the metal bounded by a 1t dimension on each side of the centerline of the weld where t is the thickness of the weld joint. The examination shall be in accord with Code paragraph N-625.

The examination method employed shall provide a means for producing a permanent record, properly identified with respect to the location and extent of the areas of the vessel examined, including annotated interpretations of all significant indications observed. The records shall be appropriate to serve as a reference examination for comparison with future inspections which may be required during the vessel's service life.

In areas where either the interpretations of the reflections observed are in doubt, or the recorded indications appear to exceed the acceptance standards of Code paragraph N-625.3, a supplemental examination shall be performed by means of a radiographic examination in accord with Code paragraph N-624.

The radiographic examination shall employ special techniques of exposure orientation necessary to fully define and interpret the reflectors observed by the ultrasonic examination.

Any indications revealed by the radiographic examination which exceed the acceptance standard of N-625.3 shall be subject to review and evaluation by the vessel manufacturer, the professional engineers who certified the vessel Design Specification and the Stress Report, and the vessel owner or its agent, before the need for repairs and retests is established.

Any weld repairs shall be performed in accord with Code paragraph N-625.4. All areas of the vessel of questionable interpretation, weld repaired areas, and the result of nondestructive reexaminations shall be recorded as part of the Vessel Fabrication Report and accompanied by appropriate identification and location of these areas on vessel drawings. These records shall be made available to the vessel owner or its agent upon request.

This requirement supplements Code paragraph N-618.

> Explanation - In recognition of the numerous fabrication processes through which a nuclear vessel must pass, each of which may cumulatively contribute to the development of flaws during the course of fabrication, a program of final inspection and examination is necessary to verify the "as built" structural integrity, prior to its acceptance for service. To obtain meaningful results, such examinations must be performed following completion of fabrication, heat treatment and testing.
>
> The examination results assume additional values in that they provide a reference for reexamination of the vessel as may be required periodically to reverify its structural integrity for continued service. The maximum advantage of nondestructive techniques employed in the vessel inspection can be obtained by comparing results between the vessel's preoperational and postoperational examinations.

§1.51 Nondestructive Examination and Responsibilities.

(a) Radiograph Examination of Welded Joints - Radiograph examination employed for the examination of welding to meet Code requirements shall conform, as a minimum, with the applicable requirements of Code Appendix IX -

Section IX - 200 Quality Control System Requirements and Section IX - 300 Nondestructive Methods of Examination, and the requirements contained herein.

(b) Nondestructive Examinations - Nondestructive examinations employed for the examination of materials and welding to meet Code requirements shall conform as a minimum with the requirements of Code Appendix IX - Section IX - 200 Quality Control System Requirements and Section IX - 300 Nondestructive Methods of Examination.

Each written procedure of (a) and (b) shall be qualified by the vessel manufacturer and made part of the quality assurance program and shall be made available for review and approval by the vessel owner or its agent. Such approval shall not relieve the vessel manufacturer of its responsibilities for compliance with Code rules.

This requirement supplements Code paragraph N-611.1.

Explanation - The structural integrity built into nuclear vessels is dependent upon the meaningful performance of the nondestructive examinations of the vessel as a means to achieve and control the quality standards during the course of vessel manufacture.

Equally important to the attainment of high quality standards in nuclear vessel manufacture is the quality assurance effort invested in establishing the adequacy of nondestructive examination methods. The vessel owner who ultimately assumes the responsibility for the safety of nuclear power plant, in which nuclear vessels represent major components, must assure himself of the adequacy of the vessel manufacturer's methods of nondestructive examination by a review of the quality assurance program.

TESTING

§1.60 Hydrostatic Testing Requirements

(a) Examination for Leakage During Hydrostatic Test. Any indication of leakage in the pressure boundary of a vessel at other than a flanged connection shall be reported to the vessel owner or its agent before corrective action is taken. Both the location and extent of the leak indication and the corrective action taken shall be reported in the Vessel Fabrication Report.

(b) Testing Temperature. Prior to and during the performance of the hydrostatic test, the vessel material temperature shall be not less than 60 F above the highest of the impact test temperatures required to meet the impact test values in Code Table N-421, taking into account materials and welds of the vessel's pressure boundary and the materials of nonpressure parts directly welded to either the inside or outside surfaces of the vessel. The test temperature shall be reported in the vessel Fabrication Report.

(c) Water and Cleaning Requirements for Testing. Prior to hydrostatic testing, the vessel interior surfaces shall be cleaned with compounds free of halogen or other deleterious material, as approved by the vessel owner or its agent.

For vessels constructed or clad with austenitic or Ni-Cr-Fe alloy, the water used for a hydrostatic test conducted below 150 F shall be demineralized water with a maximum chloride plus fluoride ion content of 25 ppm. For vessels constructed or clad with austenitic alloy, tests conducted above 150 F, but not exceeding 200 F, demineralized water with maximum chloride plus fluoride content not in excess of 1 ppm shall be employed. For vessels constructed or clad with

Ni-Cr-Fe alloy, water with an initial halogen ion content not in excess of 25 ppm may be used for tests conducted above 150 F but not exceeding 200 F provided a flush or rinse with 1 ppm demineralized water is performed. The measurements of water chemistry during the test shall be recorded in the Vessel Fabrication Report.

Following the hydrostatic test, the vessel surfaces shall be completely dried and protected from contamination by sealing all openings and using desiccants or heated dry air when practical to preclude moisture accumulation within the vessel. Such protection shall be effective until final installation of vessel in a closed system.

Requirements (a), (b) and (c) supplement Code paragraphs N-714.3, N-714.4 and N-714.5 respectively.

> Explanation - The performance of a hydrostatic pressure test upon completion of vessel fabrication serves to detect manufacturing flaws as indicated by water leaks, to locate inadequate design as evidenced by excessive distortions under pressure, and to verify the adequacy of the ductile-brittle transition properties of all materials used in the construction of the vessel.
>
> Hydrostatic testing under improperly controlled conditions may potentially injure the vessel or may expose the vessel material to deleterious effects of halogens in the test water. These effects may not be readily detected until after a significant service period.
>
> Appropriate test conditions are essential to preclude effects detrimental to the vessel's safety in service.

NEW MATERIALS--SPECIAL REQUIREMENTS

§1.70 Approval of New Materials for ASME Code-Constructed Nuclear Vessels.

(a) Crack-Susceptibility Test for Ferritic Materials - In addition to Code requirements, new materials intended for reactor vessels shall be subjected to crack-susceptibility tests meeting with the following requirements:

For each product form used in reactor vessels, the base metal, weld metal, and heat-affected zone shall be subjected to comparative tests to determine its crack-susceptibility from embrittlement by strain aging, hydrogen embrittlement, or temper embrittlement.

(1) The strain aging test shall consist of a reverse bend test, using a specimen 0.75 x 10 inch long, prestained by an initial 180 degree bend, aging at 300 F (or 550 F) for 1-1/2 hours, and finally opening the bend specimen at room temperature. The results, in terms of observed cracking, shall be compared with that of Code-approved materials of known low susceptibility to strain aging.

(2) A test to detect susceptibility to hydrogen embrittlement shall consist of immersing a precracked specimen (cantilever bar) in an electrolytic solution conducive to hydrogen release or exchange for a specified period. The results in terms of increased crack growth or development shall be compared with that of Code-approved materials having low susceptibility to hydrogen embrittlement.

(3) A test to detect susceptibility to temper embrittlement in consequence of heat treatments shall be conducted by subjecting two test plates - one plate to the postweld heat treating temperature and time as specified in the Code and

a second plate to a temperature of 900 F for 24 hours, followed by furnace cooling. Charpy-V-notch tests shall then be performed with at least 12 specimens to develop the transition curve for each test plate and to determine the difference in transition temperature shift caused by the heat treatments.

(b) Fracture-Toughness Properties - For ferritic materials intended to be used in reactor vessel region directly surrounding the core, appropriate tests shall be made to compare the fracture-toughness properties with steels of acceptable toughness.

Information on the fracture-toughness properties of the material shall be developed by an acceptable test procedure. In addition, a crack growth rate test under cyclic loading conditions shall be conducted on specimens to obtain information in the temperature range of interest.

Similar information shall also be furnished for the welded specimens of these materials to demonstrate that comparable properties are attainable in the heat-affected weld zone and weld metal as in the base metal.

Requirements (a) and (b) supplement Code Appendix VIII-100(c) and (d), respectively.

> Explanation - The application of new steels in reactor vessels with physical and mechanical properties substantially different from those currently in use is beset with some uncertainties in regard to their suitability under the long-term service effects of an irradiation environment and exposure to reactor coolant conditions.
>
> Unless appropriate tests are performed to compare the crack-susceptibility and fracture-toughness properties of new steels with those of proven and acceptable properties, the assurance of reactor vessels meeting safety requirements for extended service periods cannot be established.

AEC DEVELOPS SUPPLEMENTARY CRITERIA FOR NUCLEAR PRESSURE VESSELS

The Atomic Energy Commission is making available to industry code groups and to others in the nuclear industry tentative supplementary criteria for the design, fabrication, and inspection of pressure vessels for licensed nuclear power reactors. These vessels contain the reactor fuel and the coolant. The purpose of these criteria is to help assure that these vessels are built to the highest quality standards practicable.

The criteria have been developed by the AEC's Regulatory Staff with the cooperation of the Commission's Reactor Development and Technology staff and national laboratories. The Regulatory Staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards in the development of the criteria, and they reflect ACRS review and comment. Industry code groups have also taken steps over the past several months to upgrade existing codes and standards applicable to nuclear pressure vessels, and are actively considering further steps in this direction. These criteria may be useful to these groups.

It has been the practice of the AEC to require on a case-by-case basis that nuclear pressure vessels be designed to quality standards which supplement those presently specified by industrial codes. The supplementary criteria reflect to a considerable degree current practice in the design, fabrication, and inspection of pressure vessels for water-cooled power reactors, but are considered to be generally applicable to pressure vessels of other power reactors as well. They are intended to be used in conjunction with and as supplements to the American Society of Mechanical Engineers code rules, the specifications of the American Society of Testing Materials, and the standards of other code groups. Pending further development, these criteria are expected to be useful to pressure vessel manufacturers and to the nuclear industry as interim guidance concerning AEC regulatory requirements for nuclear pressure vessels.

Copies of the tentative criteria are available for inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and may be obtained by writing to the Director, Division of Reactor Standards, U.S. Atomic Energy Commission, Washington, D.C. 20545. Comments on the criteria may be sent to the Director of Regulation, U.S. Atomic Energy Commission, Washington, D.C. 20545.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Committee Meeting Minutes. Numbers 615 - 624

January 8-10, 1970 - January 31, 1970

ACRS-0620

NDT/PRESSURE VESSELS/CONTAINMENT SUBCOMMITTEE MEETING

WASHINGTON, D. C.

JANUARY 22, 1970

The NDT/Pressure Vessels/Containment Subcommittee met to review three items provided by the Division of Reactor Standards in a November 4, 1969 letter to the Chairman of the ACRS. These were "Criteria for Ferritic Material Fracture Toughness Requirements", "Technical Bases for Development of Material Fracture Toughness Criteria", and "Material Surveillance Program". Present at this meeting were the following:

ACRS

H. Etherington
S. H. Bush
D. Okrent
W. E. Cooper, Consultant
H. T. Corten, Consultant
M. C. Gaske, Staff

Div. of Reactor Standards

E. G. Case
R. R. Maccary
S. S. Pawlicki

Division of Reactor Licensing

R. C. DeYoung
A. W. Dromerick
W. R. Johnson
J. P. Knight
K. R. Wichman

Executive Session

1 inquired as to whether it was thought that the "Criteria for Ferritic Material Fracture Toughness Requirements" are generally acceptable. 1 indicated that he had fundamental questions regarding the requirements and did not know whether they are acceptable or not. 1 reported that work has been carried out on the requirements over the last 2-1/2 years and that they represent a major deviation from the ASME requirements. 1 suggested that, if one is willing to accept the criteria, one should be willing to accept the bases for the criteria although there are some gaps in the information provided in the bases. 1 indicated that the draft of "Criteria for Ferritic Material Fracture Toughness Requirements" had been reviewed as a part of the HSST planning program and comments had been given. He said that he, and probably others who had reviewed the criteria, had given basic comments which had not been incorporated into the draft of the criteria.

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Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE 1964 O-214-629 [redacted] said that he did not see how much more could be done in the immediate future than to use Charpy V-notch testing. [redacted] agreed with this. [redacted] stated that the criteria are tied heavily to the Charpy tests. [redacted] indicated that he does not believe it is possible to assign a failure probability, such as 10^{-6} per year, to the probability of vessel failure that might be associated with use of the proposed fracture toughness requirements. He thought that the value was not that small and reported that incorrect steel has been used in non-nuclear vessels which has led to the failure of such a vessel every couple of years. [redacted] thought that, if such a mistake were made, weaker material would certainly be used and the fact detected during the hydro test. [redacted] indicated that, if stronger steels were used, difficulty might be encountered with the welds.

[redacted] believed that the document regarding the bases of the fracture toughness requirements should not be published. [redacted] indicated that the document raises a number of questions, and [redacted] stated that he did not believe it was well written.

[redacted] stated that the question should be asked as to what level of protection one is trying to accomplish with the criteria and whether the criteria accomplish this. [redacted] suggested that the wrong welding material might be used in fabrication of a pressure vessel. He said that many welds will not be inspected or only partially inspected during the life of a plant. There are questions as to how well the neutron flux and the temperature at the pressure vessel wall will be known.

[redacted] indicated that an attempt has been made in other areas to obtain information regarding the probability of failures and that these efforts have been unsuccessful. [redacted] stated that the proposed requirements would represent what the AEC says is sufficient. He indicated that it is not obvious that a requirement of a minimum Charpy V-notch fracture energy of 50 ft-lbs would place material in Pellini's plastic enclave.

[redacted] drew a graph indicating that, in terms of ft-lbs to cause fracture during a Charpy V-notch test, specimens taken in the transverse direction have lower values than those taken in the longitudinal direction. A minimum Charpy V-notch fracture energy of 50 ft-lbs would be required for material throughout the pressure vessel life. Linear fracture mechanics techniques can be used for evaluation of material having a Charpy V-notch fracture energy of 50 ft-lbs or less. A Charpy V-notch value of 50 ft-lbs corresponds to a k_{Ic} value of approximately 150. [redacted] indicated that the size flaw which corresponds to k_{Ic} value of 150 is quite large and that he feels fairly comfortable regarding the use of this value. [redacted] said that when he sees that a flaw has to be of a size in the 8" to 10" range to cause difficulty he is not concerned. [redacted] indicated that the k_{Ic} value of 150 would not lead to crack propagation for a crack of a size up to approximately 6 to 7 inches deep and 15 inches long.

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[redacted] pointed out that the criteria do not restrict themselves to being applicable only to presently used steels. He inquired as to whether it was acceptable to use a steel with an initial Charpy V-notch value of 51 ft-lbs at operating temperature. [redacted] stated that, if there is to be a degradation of a steel due to irradiation, then the original fracture energy would have to be higher. [redacted] indicated that he is seeking a failure rate of less than 10^{-8} per reactor-year. [redacted] thought that, in establishing the probability value for pressure vessel failure, there would need to be probabilities established regarding the use of the wrong pressure vessel wall material or incorrect welding rods.

[redacted] thought that with the pressures and temperatures which will be present in a reactor that cracks would penetrate the pressure vessel wall and result in leakage before the vessel would break. [redacted] believed that the situation might be a bit marginal and commented on the uncertainties in knowing the temperature of the radiation samples vs. that of the pressure vessel wall and the differences in the fluxes to which each is subjected. [redacted] pointed out that the minimum fracture energy of 50 ft-lbs is for the inner surface of the pressure vessel and that higher values would be present through the pressure vessel wall. He stated that a value of 50 ft-lbs had been chosen based on reviewing the Pellini data which indicated that ductile tear could occur in materials having fracture energy of approximately 25 to 30 ft-lbs. [redacted] said that the ASME code does have some requirements regarding fracture energy. [redacted] stated that it does but that these requirements are only for longitudinal samples and a value of only 30 ft-lbs is required.

[redacted] pointed out the following regarding the probability of a pressure vessel failing which had been built using the proposed criteria.

1. The belt region consists of a very simple geometry, a right cylinder.

2. Chemistry testing will be required for all the plates.

3. Sampling and testing of the weld material will be required.

4. The inner surface at the belt region will have a lower fracture energy than the outer wall.

5. The pressure vessels will have an elevated temperature before a high pressure is reached.

[redacted] said that the belt region is not the area where he would expect a leak to occur, but he does expect that leaks will occur in nozzle regions.

[redacted] read a newspaper article regarding a failure in an F-111 aircraft. It was reported that the failure was either due to poor inspection, the inspection being concentrated too much in a region where flaws were expected, or the inspection procedures not being sensitive enough.

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[redacted] stated that the irradiation surveillance program is a long term problem. He indicated that, when the present day research program is completed, different criteria may be needed. [redacted] said there is a need to obtain data from the HSST program and to reach a consensus regarding the meaning of the data. It was reported that six pressure vessels with 6-inch walls have been purchased as a part of the HSST program. The first of these is to be delivered this fiscal year, and testing of the vessel will be begun in approximately 18 months.

Regulatory Staff

Mr. Pawlicki briefly reviewed the contents of the document "Technical Bases for Development of Material Fracture Toughness Criteria". The k_{Ic} data have been found to indicate a sharp increase in the k_{Ic} values at the point where the ductile-brittle transition point occurs. There had originally been fear that the k_{Ic} values would not increase rapidly at this point. Mr. Pawlicki said that it is proposed that operation always be in upper region of the transition range. Dr. Okrent inquired whether operation will always be in the fully-plastic enclave. Mr. Pawlicki said that the criteria are not written in terms of the enclaves. He reported, however, that in 99% of the cases operation will be in the plastic enclave.

Dr. Okrent inquired as to whether a minimum ~~operating~~ temperature of 120°F is presently being added to the NDT temperature. Mr. Maccary reported that such a criterion has not been required by the Regulatory Staff. The ASME code requires either the use of the dropweight test or the Charpy V-notch test. NRL research indicates that the Charpy V-notch test may not provide adequate data. In one case, the NDT temperature established by the dynamic tear test was at approximately 120 ft-lbs and there was a gross lack of correlation with the Charpy test. Dr. Cooper stated that, in this particular case, there is a question of the validity of the dynamic tear test for the particular material used. Dr. Bush said that this is the reason that the use of A-533 requires considerable documentation. It was reported that the transition temperature determined by the dynamic tear test will be used if the information is available and the value obtained is higher than that from the Charpy V-notch test. It was reported that General Electric requires both tests for the pressure vessel material in the core region.

Mr. Pawlicki stated that the use of the NDT temperature plus a certain temperature value is confusing and has not always been conservative. Accordingly, the Regulatory Staff is recommending the use of the minimum fracture energy of 50 ft-lbs, which corresponds to a k_{Ic} value of 150. This value is in the upper range of the validity of elastic fracture mechanics.

Dr. Okrent recalled the Staff statement that the material would not necessarily be required to operate within Pellini's plastic enclave. Dr. Okrent commented that the material may be on the borderline where it would have more of a brittle nature. Mr. Pawlicki replied that he believes that the minimum k_{Ic} value of 150 is adequate for a pressure vessel thickness up to 12 inches.

Mr. Corten, under the sponsorship of the HSST program, is to provide advice regarding the correlation between the Charpy V-notch values and the k_{Ic} values. Dr. Okrent pointed out that the criteria are written for ferritic material and that the Regulatory Staff had found in one case that the dropweight test was not adequate. Mr. Pawlicki stated that the Regulatory Staff believes that the Charpy V-notch test results should be verified by dropweight tests and the most conservative values used. Mr. Etherington said that there is a question as to whether the criteria should be restricted to use for presently used steel. Mr. Pawlicki replied in the affirmative. Mr. Maccary said that the Regulatory Staff has not examined all ferritic materials but he believes that the proposed criteria are in the right direction. He stated that the question might be raised as to whether the criteria are sufficiently conservative and that the Regulatory Staff will continue to examine to which ferritic materials the criteria should apply. The criteria will be revised to specify the applicable range of ferritic materials.

Dr. Okrent inquired as to what degree of assurance of pressure vessel integrity should be attained and what degree of protection of the public should be achieved. Mr. Maccary replied that the criteria are such that the fracture toughness indicates that a through wall flaw can occur and there still be a degree of safety for continued operation. Dr. Okrent inquired whether Mr. Maccary was saying that the probability of the health and safety of the public being in danger is zero. Mr. Maccary indicated that, with the proposed limit on fracture toughness, there should be reasonable assurance of no propensity for rapid crack propagation. Dr. Okrent inquired regarding what level of assurance regarding vessel integrity should be provided. Mr. Maccary replied that, from the data available, he believes there would be zero probability of pressure vessel failure. He indicated, however, that new data might be developed which would indicate otherwise. He thought that the Regulatory Staff should strive for the zero probability value. Mr. Corten said that, if the proposed criteria are met, there should be zero probability. He thought that this may not be attained, however, because of failures in the inspection process. Dr. Okrent inquired as to how small flaws have to be before one need not worry about them. Mr. Corten replied that a way to put a probability on this has not been found. It was stated that Mr. Pellini has urged that considerable work be done regarding weld material and heat affected zones. Dr. Cooper indicated that unirradiated weld material is usually better than base material. Dr. Bush said that considerable data are available regarding submerged arc welding. Dr. Okrent pointed out that there was some thought that there might be a plateau for the k_{Ic} values, but then it was recently found the curve turns up with temperature.

Dr. Cooper indicated that he would like to see a move away from the use of the Charpy V-notch test, as this would get away from the problem of from which direction a sample is taken. It was reported that the specifications for each reactor pressure vessel require that a review be made by the purchaser of where the irradiation samples are to be taken.

Mr. Corten stated that higher minimum values should be specified for the material that will not be irradiated than for that which will be irradiated. He indicated that the stress calculations are the most difficult in the nozzle and bolted-on head regions.

Dr. Okrent suggested that, perhaps, annealing might be required to stay above a certain value. Dr. Bush said that annealing may result in degradation of the primary system. He indicated that, with a fracture energy of 70 ft-lbs, a flaw approximately 4 feet long would be necessary in order to cause a problem. Annealing three or four times during the life of the vessel might be less conservative than permitting the critical flaw size to decrease to 1-1/2 or 2 feet long without annealing.

Whenever the minimum fracture energy levels cannot be met, fracture mechanics may be applied to justify lower values. Mr. Etherington stated that, if 50 ft-lbs is an appropriate requirement for irradiated material at the belt region, then a higher value should be present in the nozzle region. Dr. Bush believed it highly improbable that different materials would be used in fabricating different parts of the pressure vessel. Mr. Corten thought that a single criterion should be used and that, perhaps, the material be required to have the same initial fracture energy value throughout the vessel. Dr. Cooper stated that the criteria invite the use of linear fracture mechanics techniques and that he was worried about the possibility of dynamic tear. He said that the true upper shelf should not be allowed to fall below 35 to 40 ft-lbs. He suggested that new criteria might be added that the upper shelf must have a fracture energy of at least 70 ft-lbs initially and at least 35 to 40 ft-lbs at the end of vessel life. Mr. Maccary said that he did not know whether operation with a material having a fracture energy less than 50 ft-lbs can be justified.

Mr. Etherington inquired whether the Regulatory Staff had obtained comments regarding the proposed criteria. Mr. Maccary reported that they had from five different individuals or groups. He said that the Regulatory Staff does not plan to publish the bases document and that this document has been generated as a result of a Subcommittee request. Mr. Case has indicated there is a question as to how the criteria should be published.

Mr. Maccary agreed that the "Criteria for Ferritic Material Fracture Toughness Requirements" will be changed to restrict the type of ferritic material to which they are applicable. Mr. Etherington questioned the use of the word "Mechanical" in the title "Criteria for Ferritic Material Fracture Toughness Requirements of Components and Piping of Mechanical Fluid Systems". He also questioned the use of the word "within" in the first sentence of the criteria. The phrase, "within the reactor coolant pressure boundary" can be construed to mean core internals, etc. Dr. Cooper suggested that the phrase "pressure retaining" might be used in the criteria. Dr. Bush indicated he was concerned that the criteria might appear to apply to bolts. Dr. Cooper suggested that the criteria might exclude bolting material. Dr. Okrent commented that, in writing criteria, one usually asks for conservative rather than adequate requirements. Dr. Bush pointed out that the statement that no impact testing shall be required for

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ferritic materials 1/2 inch and less in thickness would have excluded such testing for the Saxton pressure vessel which was laminated. Mr. Etherington commented that the word "normalizing" in section (c) (1) needs to be clarified.

Dr. Okrent recalled that Mr. DiNunno used to suggest the use of the NDT temperature plus 120° F where dynamic loading may be present. Mr. Maccary said that such a criterion may not be conservative in that the transition curve may be very shallow and that 250° F might need to be added rather than 120° F. He said that this was the reason that the decision was made to state a minimum fracture energy. Dr. Bush said that, in the most degraded condition allowed by the criteria, the material would be on the borderline of the plastic enclave. Dr. Okrent said that Mr. Pellini had stated that operation should be carried out only in the plastic enclave. He inquired whether there was any reason that this could not be done. He pointed out that the use of NDT plus 60° F was thought to be acceptable a few years ago and now it is being stated that this criterion is not conservative. Dr. Okrent inquired as to the reason why a minimum fracture energy of 60 or 65 ft-lbs should not be required. Mr. Maccary stated that, perhaps, a minimum upper shelf energy level should be established. Dr. Bush said that he would like to use the 50 ft-lb value and would be willing to have the value changed if the HSST program does not indicate that this is a very safe value.

Mr. Corten said that, with the proposed criteria, a pressure vessel could have a crack at the end of the 40-year life which is completely through the vessel wall and two times the wall thickness in length and still not run. Mr. Pawlicki indicated that he did not expect that a crack through the pressure vessel wall would necessarily let water out. Mr. Corten stated, however, that one could not possibly miss finding such a crack during an inspection. Dr. Bush believed that such a crack could be detected by humidity or radiation monitors or by other means before it would run. Dr. Okrent pointed out that, until recently, it might have been said that turbines never fail at nuclear facilities; however, recently one failed. Mr. Corten suggested that, instead of increasing the minimum fracture energy, better inspection methods might be used. Dr. Okrent predicted that flaws will be found and licensees will wish to continue operation using the defective pressure vessel. Mr. Cooper questioned as to what mechanism might cause a flaw to grow in the beltline region.

Dr. Okrent stated that, if it can be said on a solid basis there is zero probability of failure with the use of a minimum fracture energy of 50 ft-lbs, then the use of such a value is acceptable. He indicated that, if this cannot be said, then the question should be raised as to why a more conservative minimum fracture energy is not used. Dr. Okrent inquired regarding the reason for not using a higher number. Mr. Etherington stated that one reason a higher value might not be acceptable is that it might result in rejection of good plate material. Mr. Pawlicki indicated that there might be difficulty in dealing with industry in convincing them that even a 50 ft-lb value is acceptable. Dr. Cooper said that use of too high a minimum value would force the use of annealing. Dr. Okrent indicated that the ACRS had been told that annealing can be performed.

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[illegible]

Mr. Pawlicki said that applicants agreed that they will be prepared to anneal because they do not believe they ever will have to do so.

Mr. Maccary stated that the ASTM specifications set forth the requirements regarding samples taken from a heat affected zone. Section (d)(3) on page 4 of the "Fracture Toughness Requirements" is to be changed to specify a value of 25% in the statement "the transition temperature . . . shall not be greater than the minimum operating temperature at which the pressure exceeds 20% of system operating pressure . . .". Dr. Cooper said that he did not understand how the transition temperature is obtained from the dynamic tear test. Mr. Pawlicki suggested that the criteria could leave an option of using something other than the Charpy test results and providing a correlation with the Charpy values. Dr. Bush suggested that such an option be given but subject to the approval of the procedure by the Regulatory Staff.

Dr. Okrent questioned the use of the value of 250° F instead of a lower value in (4) on page 6. Dr. Okrent suggested that such problems as possible flux depression in capsules and other difficulties regarding correlation of sample data to pressure vessel material data might be indicated in the criteria. Mr. Etherington thought that temperature differences between the capsules and the pressure vessel wall might not represent much of a problem. Approximately six neutron energy threshold monitors are used to determine the neutron irradiation of a capsule. The use of the value of 250° F instead of a lower value on pages 2 and 3 of "Material Surveillance Program" was questioned. Dr. Okrent referred to the last statement on page 3 of the requirements regarding the "Material Surveillance Program". He said that it was difficult to believe that different reactors will be operated under comparable conditions and service. Dr. Okrent thought that, even at the same site, reactors will be operated differently. He indicated, also, that he had the feeling that the operator should be responsible for a surveillance program and was surprised to see the phrase "supplied to one or more applicants" stated in (c) on page 3.

Dr. Bush pointed out that Charpy testing machines often malfunction so as to make the impact values high. He suggested that the requirements might indicate that the Charpy testing machine must be calibrated by appropriate procedures. Dr. Okrent inquired regarding how the Charpy data are used to draw a curve. It was reported that this was done by "Eyeball". Dr. Okrent questioned whether there should be a requirement that the most conservative values be used. Mr. Maccary indicated he believed that there are ASTM requirements regarding how the data must be plotted. Dr. Okrent inquired as to whether persons performing the Charpy tests need to be qualified. Apparently, little if any qualification, other than a familiarity with ASTM 185, is required. Mr. Etherington stated he believed that the Subcommittee did not have any major reservations regarding the requirements of "Material Surveillance Program". The Regulatory Staff is to revise "Criteria for Ferritic Material Fracture Toughness Requirements". Mr. Etherington inquired as to how soon this might be done, and Mr. Maccary indicated that this should be completed around the time of the March ACRS meeting.

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Mr. Etherington asked regarding the Regulatory Staff's position relative to annealing. Mr. Maccary stated that there are limited data regarding the components that would be subjected to annealing. There are questions such as the effects on the tube-to-tube-sheet joints. Mr. Maccary indicated that the Regulatory Staff is reluctant to consider annealing at temperatures above the design value of 650° F. Dr. Okrent inquired whether it is practical to anneal at 650° F. Mr. Maccary replied that he believed it has been verified that the PWR pressurizer heater and pumps can create a temperature of 650° F. Mr. Etherington inquired about strain aging and temper embrittlement. It was reported that these are items that will be investigated as part of the HSST program. Dr. Bush indicated that these represent potential problems to materials not now being used in reactors.

Dr. Okrent reported that he had asked an applicant whether he could state that his PWR system could be annealed at 650° F without going back and surveying to make sure this could be done, and the applicant stated unequivocally that he could. Dr. Okrent indicated that the Regulatory Staff might review the problems regarding annealing. Mr. Maccary said that the Regulatory Staff is thinking of asking applicants how they would anneal if this were found to be necessary.

Dr. Okrent stated that some applicants have said that the startup period is the most dangerous time regarding the possibility of failure of the pressure vessel. Dr. Bush said he assumed that the applicant would be required to stay within specified technical specifications limits regarding temperature and pressure values during this time.

Dr. Bush pointed out that there is a two-day meeting regarding the HSST program and that the meeting will be held on March 31 and April 1, 1970 at Oak Ridge. He indicated that this meeting would represent a good opportunity for anyone interested to remain current regarding the HSST program.

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April 7, 1970

H. Etherington, Chairman
NDT/Pressure Vessels/Containment
Subcommittee

MINUTES OF MEETING OF APRIL 2, 1970

Minutes of the meeting held in the Alexander Motor Inn, Oak Ridge, on April 2, 1970, are attached for your review. A copy has been sent to each of the remaining ACRS members. Please advise me regarding whether or not our consultants should receive them.

I have sent Dr. Mager of Westinghouse a list of names and addresses of meeting attendees so he can forward a summary of what he presented.

J. E. Hard
Senior Staff Assistant

Attachment:
Minutes of Meeting of April 2, 1970, Oak Ridge, Tennessee

cc: Remainder ACRS Members, w/attachment

FILE: NDT/PV/Containment RD9-4

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MINUTES OF NDT/PRESSURE VESSELS/CONTAINMENT SUBCOMMITTEE MEETING OAK RIDGE, TENNESSEE APRIL 2, 1970

Summary

The Subcommittee met with the Regulatory Staff and ACRS consultants to discuss the following proposed criteria:

Appendix F, Reactor Material Surveillance Program Requirements, February 20, 1970

10 CFR 50.55a(i), Fracture Toughness Criteria, March 18, 1970

Appendix F was approved for incorporation of the day's comments and for final drafting.

Fracture Toughness Criteria was sent back to the Staff for study and revision. The effect of irradiation on fracture resistance of thick reactor vessel walls continues to be a major concern.

A Westinghouse representative (T. Mager) presented fracture toughness data which indicated that K_{Ic} may exhibit a shelf behavior with increasing temperature.

Attendance

ACRS

H. Etherington
S. Bush
D. Okrent
J. Hard, Staff
W. Cooper, consultant
H. Corten, consultant
F. Loss, consultant
P. Paris

DRL

R. Birkel
V. Benaroya

DRS

R. Maccary
S. Pawlicki
J. Knight
K. Wichman
P. Norian

RDT

H. Behrman
J. Hunter

ORNL

F. Witt

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Meeting with All Attendees

Mr. Etherington requested Mr. Witt's comments on the NDT requirements portion of the draft 50.55a(i), Fracture Toughness Criteria. Witt recommended that NDT + 60°F be used as the minimum service temperature criterion.

Dr. Okrent asked Mr. Witt his opinion on annealing requirements. Witt felt that, if annealing is required, surveillance specimens must be provided to judge when this should be done.

Maccary passed out a set of curves which are examples of how the criterion would work in practice. Indian Point 3 was the specific project involved. The rolling ratio for these plates is about 1/2. Figure 3 of the attachment shows the application of the fracture toughness criteria to the IP-3 beltline materials. Use of the proposed criteria would result in a minimum service temperature of 132°F vs 45°F for the present criteria.

With the proposed criteria applied to other-than-beltline materials, the minimum service temperature is much higher (175°F) as shown on Figure 5. (This is because of the lower acceptable fracture toughness away from the beltline.) During the life of the plant, the beltline requirement will gradually increase and could overtake the 175°F requirement.

For all these discussions, the transverse Charpy-V energies are assumed to be 2/3 of the longitudinal values. On the basis of these comparisons, Maccary stated that three of the existing IP-3 vessel plates would be unacceptable. Using the previous version of the proposed criteria, no plates would have been rejected.

There was some discussion on whether or not the Charpy curves are directly applicable to pressure vessels since they only describe specimen behavior. As pointed out by Mr. Corten, these data may not have direct relationship to rapid crack propagation in a vessel. Dr. Paris reinforced Corten's statement.

Mr. Etherington asked Witt if his size effects also apply to tensile specimens. Witt felt that they do. There was some Subcommittee concern that the [redacted] proposed criteria is written for 3" and thicker plates whereas the thinnest pressure vessel wall being constructed is 6-5/8" in thickness. Therefore, the criteria may not be conservative in this sense.

Dr. Bush cautioned the group about unwarranted increases in required Charpy values. In his opinion, much vessel plate may have to be discarded if the standards become too restrictive. Dr. Cooper's opinion was that experience with vessels $\angle$6" thick has been good even though the standards have been poor. Above 6", there is reason for concern because of the sizing effects.

He worries that NDT + 60°F may not be adequate for the larger sizes, and the thicker the vessel, the more the concern. Though he wouldn't worry about standing next to a thick wall being hydrotested at 100°F, he would feel better at 170°F. If the temperature went above 212°F, he wouldn't be any place around because of the safety problems with any leaking steam. Dr. Okrent felt that these bases for deciding what's safe and what isn't may not be applicable to public safety concerns. He stated his conclusion on the prior discussions, that a single criterion for thick walled vessels may not be appropriate. Dr. Bush felt that there may be inadequate data to say that the proposed criteria give safe vessels in the very thick sections.

Presentation by T. Mager, Westinghouse (All Attendees present)

Westinghouse has been looking for a fracture toughness upper shelf for thick reactor materials. Dr. Mager presented some new data from his research lab. (Mager will send copies of this data to all attendees.) He asked that the data be handled in a confidential manner. The K_{Ic} values for a NiCrMoV material showed an upper shelf behavior with temperature (previously, an upturn of the curve had been expected). The shelf behavior has been seen in other materials, too. Mager recommends that the HSST program use mild steels and evaluate the questions brought up by these discussions. He felt that a K_{Ic} value of 50 ksi $\sqrt{in}$ would be acceptable for a 1-1/2" crack and that a Charpy V value of 30 ft. lbs., post-irradiated, is probably acceptable. Westinghouse's second recommendation is to irradiate A-533 material and try to fit this information to the present data. Different steels may show different behavior than the behavior discussed above. Also, since the yield strengths tend to saturate at about $2\text{-}3 \times 10^{19}$ nvt, the K_{Ic} shelf may drop off at the higher irradiation.

[redacted]

Meeting sans F. Witt, T. Mager, and RDT Representatives

Discussion of Proposed 10 CFR 50.55a(i), Fracture Toughness Criteria (3/18/70 Draft)

Mr. Pawlicki introduced the subject for the Regulatory Staff. He observed that the geometrically scaled up specimens with scaled up cracks, will fail at lower stress than the thin ones. This must be factored into the Regulatory considerations. Pawlicki felt that the considerations fall into three categories:

1. Nominal operation of vessels - establish safety margins.
2. Anticipated transients - determine how margins are reduced.
3. Acts of God and other undefined accidents - determine how much energy would have to be absorbed for the vessel to fail.

Dr. Okrent observed that there may be two categories of anticipated transients; those with and those without protective action. Pawlicki pointed out the problem of adding another 60°F to NDT + 60°F in that, for materials with different Charpy slopes, the margins become different for the different materials.

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Mr. Etherington asked for consultants' comments on the importance of the fracture energy absorbed per unit volume of material versus size (this is Witt's concern). Dr. Cooper felt that fracture area is not the correlatable parameter nor is fracture volume. For brittle failures, the fracture energy goes up with L^2*. Mr. Loss generally would be happy with adding more margin to the NDT + 60°F, dependent on a satisfactory Charpy curve and a good correlation between K_{Ic} and Charpy results. He later added that there is a difference between flawed and unflawed behavior.

[redacted]

Dr. Paris would recommend a compromise between the pure Charpy approach and the pure K_{Ic} approach. K_{Ic}/yield strength may be a useful parameter. A limit based on yield strength vs Charpy shelf was one of Dr. Paris' suggestions.

Prof. Corten commented that a level of Charpy upper shelf energy and an NDT + some margin limit is required. Some relationship should be established between upper shelf and section thickness.

Dr. Cooper added that he agreed generally with both Paris and Corten. He stated that the average analyst would use L^3 as the correlation with fracture energy but that this would be unconservative. He was uncertain as to what value between L^2 and L^3 should be employed.

Mr. Etherington asked if a different approach should be taken on the criteria. Paris repeated that these limits should be based on thickness and on yield strength. The thicker vessel should have more restrictive limits than the thinner one. Cooper thought that the industry can't wait for more sophisticated criteria and that the proposed should be issued soon.

Dr. Okrent asked if there were any reservations about thick sections. Mr. Loss felt there should be more margin for the thinner sections. For some steels, the NDT + 60°F is not conservative because of the curve shapes and method of determining NDT point. The general subject of changing the basis of determining NDT was discussed with no conclusion. (The problem is that the DT test value falls near the toe of the Charpy curve.)

In response to Dr. Okrent's question on whether or not the present approach for thick sections was conservative, Pawlicki stated that the proposed criteria are in concert with Witt's findings. Paris stated that the situation is marginal for vessels 8" thick and larger and that a criterion without provisions for change in thickness is probably not adequate. Paris would require, for 6" thick irradiated sections, a 55 ft-lbs minimum Charpy value. For a highly irradiated 6" section, the number would be 90 ft-lbs. For a 12" unirradiated section, 57 ft-lbs is recommended, for 12" irradiation the number becomes 115. His method of analysis is conservative because he assumes uniform fluence through the vessel wall. According to Dr. Bush, if these numbers are used, building reactor vess

*L is the thickness in these discussions.

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are used, one better forget about building reactor vessels. Paris' numbers are based on fluences which get yield strengths up to 10^5 psi. Lack of K_{Ic} data in the upper shelf region continues to be a problem.

Caucus

In view of the consultants' conserns (principally Paris') about irradiation effects in thick sections, the Subcommittee was faced with a thorough revision of the proposed criteria to incorporate fracture toughness requirements in addition to NDT criteria, or with an interim issue of the proposal with a proviso on irradiation effects in the thicker sections. The latter course seemed most appropriate to the Subcommittee.

Continued Discussion of Proposed 10 CFR 50.55a(i)

Dr. Bush's comments on the draft were discussed and handed to Maccary for incorporation in the next draft (Maccary handed out a 3/18/70 draft of the criteria).

A considerable discussion was held on the number of degrees the Charpy curve should be translated /Requirement ii(a)(1)/ and on the specification on the Charpy "fix" value (Dr. Bush would recommend 30 ft-lbs as the basis here). Dr. Bush warned against increasing the numbers for minimum Charpy adjusted fracture energy because of the opposition and delays which will come from industry. Cooper asked for limits in the criteria which would prohibit shop hydro testing at $>200^{\circ}$F. Dr. Okrent expressed his difficulty in understanding Cooper's concern since the name of the game is public health and safety and not concern over the safety of shop workers.

[redacted]

Discussion of Appendix F - Surveillance Requirements

A page-by-page review was performed of this 2/20/70 draft and modifications made as shown on the marked up copy in the ACRS files. During the discussion it was brought out that double annealing of the vessel wall during vessel life would probably require a specimen capsule in addition to what is already required by Appendix F. Changes to the draft were largely editorial. DRS was to redraft this for final review.

Discussion of 50.55a(i)

A page-by-page discussion was conducted on the 3/18/70 draft (comments are included in the ACRS file copy of the marked up draft). Dr. Okrent had a concern about how system overpressure considerations are handled by the proposed criteria. Cooper pointed out that pressure containing systems are both pressure-rate sensitive and pressure-time sensitive. Flange leakage can help in limiting pressure rises. One possible solution to this overpressure question would be treatment in a separate criterion. According to Mr. Etherington, all expected pressure rises should be accommodated by the proposal. DRS was to study this further. Maccary pointed out that the proposal is being written with the thought that it will be amended as necessary, perhaps every six months. Words are to be included to this effect.

[redacted]

Prof. Corten believed that the question of whether or not Charpy tests should be conducted at 500°F should be evaluated. He was to give this question some study.

He also felt that the 50 ft-lbs adjusted fracture energy for plates, etc., was about appropriate. This gives him no trouble with an unirradiated vessel. Cooper liked 50 ft lb provided the 10 ft lb adder is increased to 15 ft lb. Dr. Okrent would require a qualifying statement for thick sections, based on any new information on effect of irradiation.

Some of the more significant changes to the March 18, 1970 draft are as follows:

1. An upper limit to tensile strength is to be added.
2. System transients are to be listed.
3. A comment is to be included on suitable operating restrictions.
4. Operation with known flaws is to be discussed.

In addition, the Staff is to review the suggestion that the following limits apply:

Thickness	Min. Adjusted Charpy
$t \geq 5$ in.	50 ft lbs
$t = 2$-5 in.	40 ft lbs
$t \leq 2$ in.	35 ft lbs

The upper shelf shall be 15 ft lbs higher than these values.

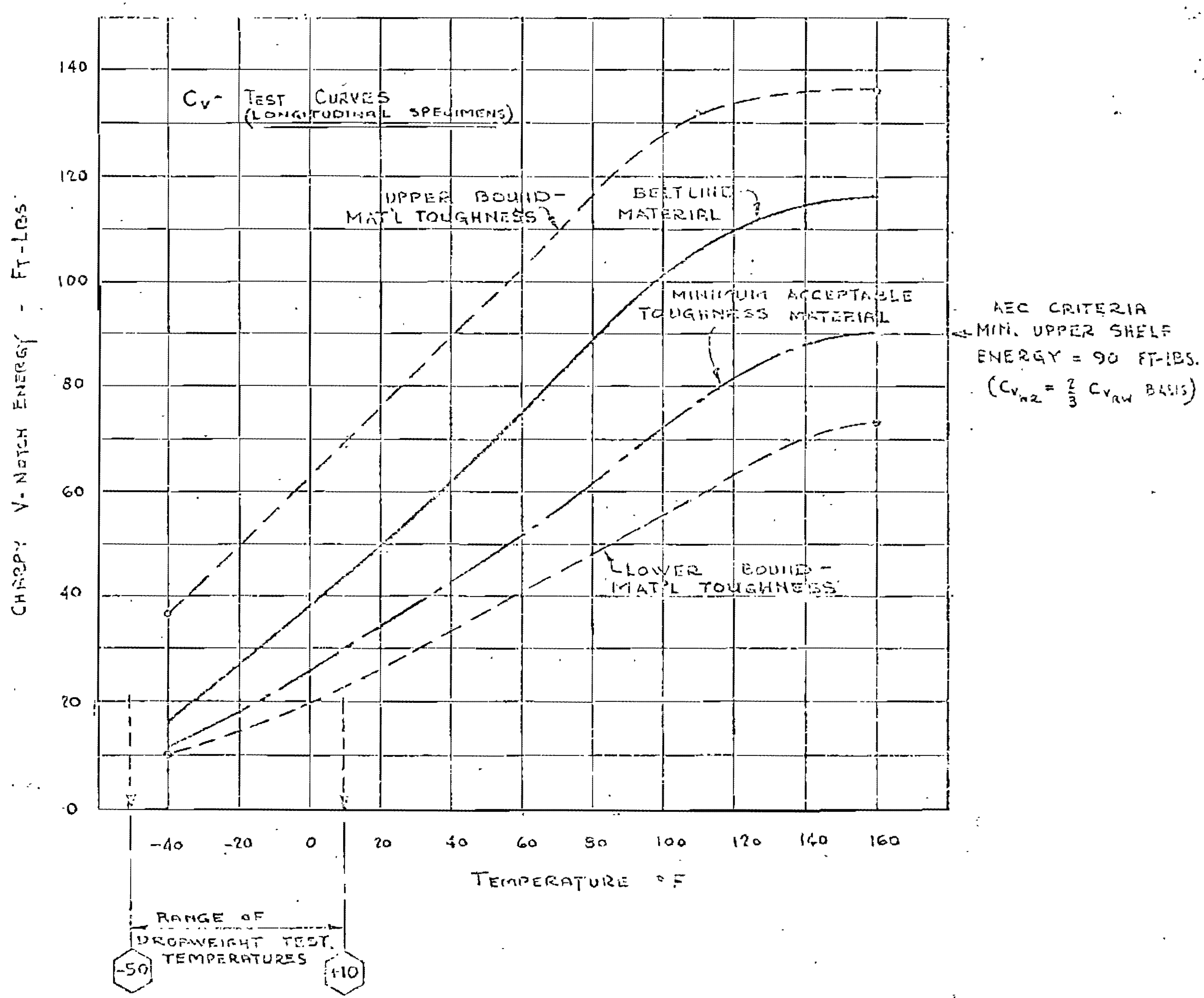
Further Subcommittee discussion will be required following Staff review and incorporation of the members' comments.

[redacted]

* * *

Attachments:
1. Charpy V-Notch - Impact Properties - Indian Point III Reactor Vessel, dtd 3/30/70
2. Use of CVN for Semi-Ductile or Non-Frangible Simple Fracture Criteria for Pressure Vessels, Paul C. Paris

CHARPY V-NOTCH - IMPACT PROPERTIES - INDIAN POINT III - REACTOR VESSEL



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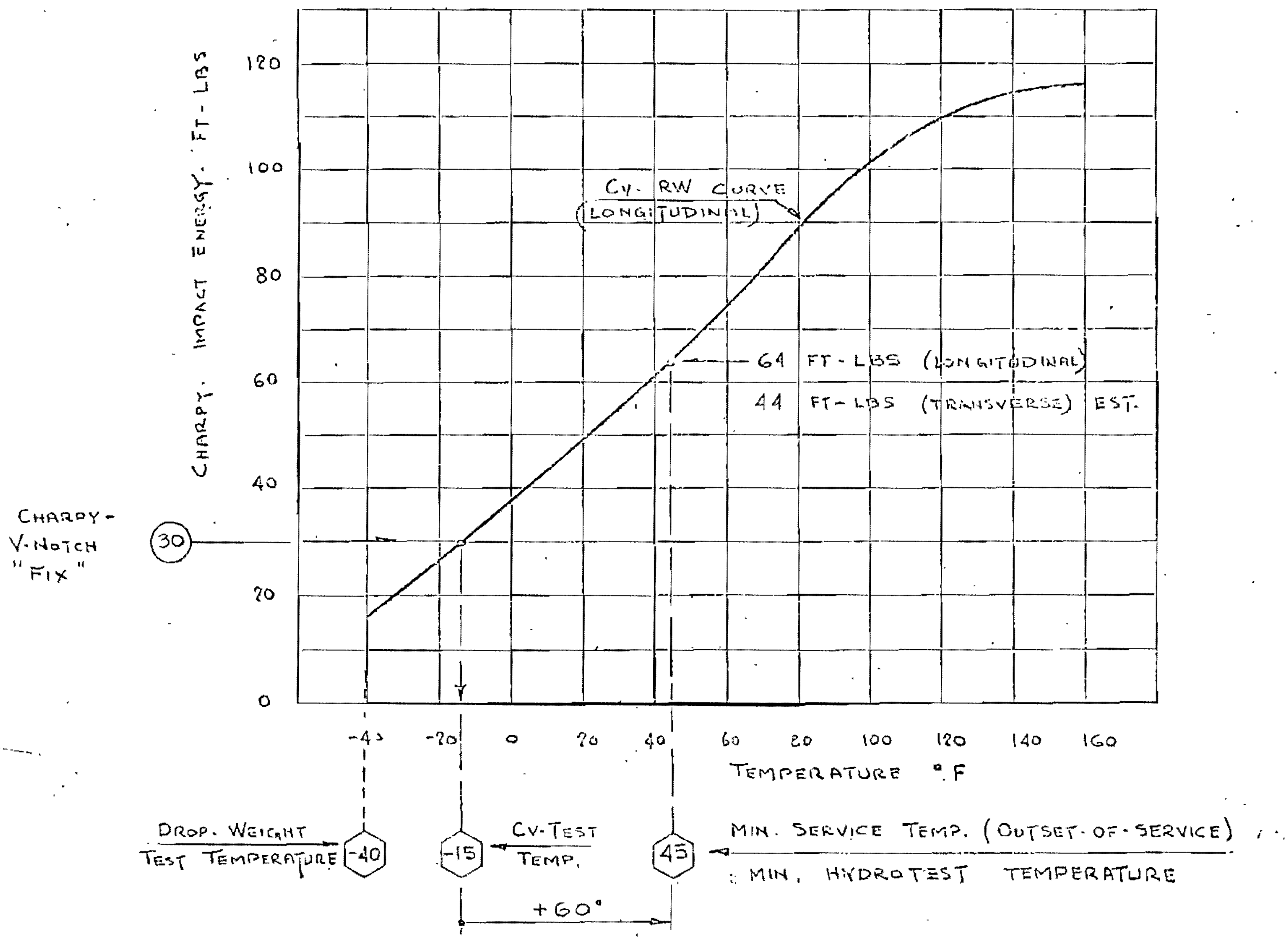
3-30-7

ASME SECTION III CODE CRITERIA

APPLIED TO

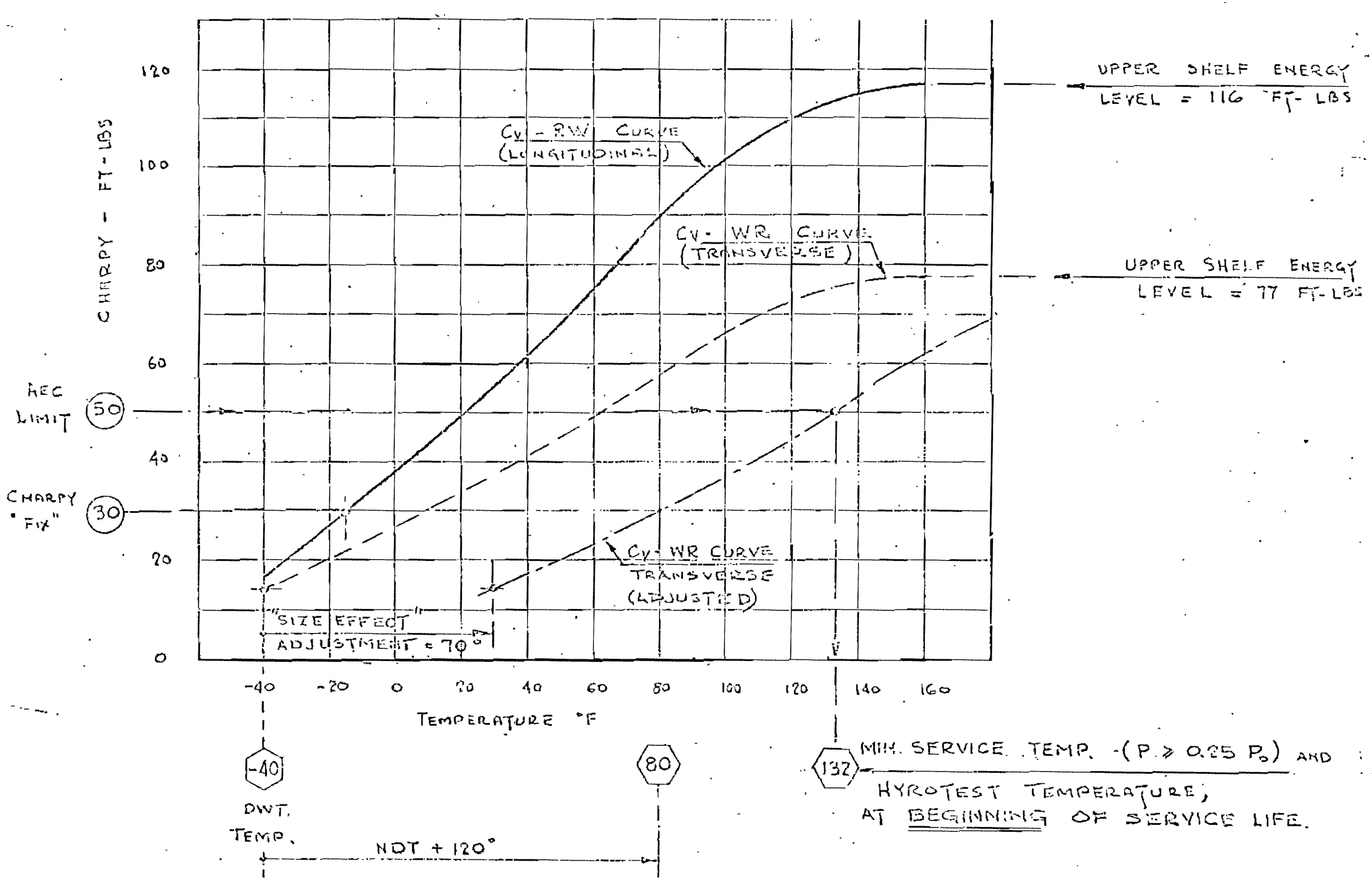
BELTLINE REGION - REACTOR VESSEL



AEC FRACTURE TOUGHNESS CRITERIA

APPLIED TO

BELT LINE REGION - REACTOR VESSEL



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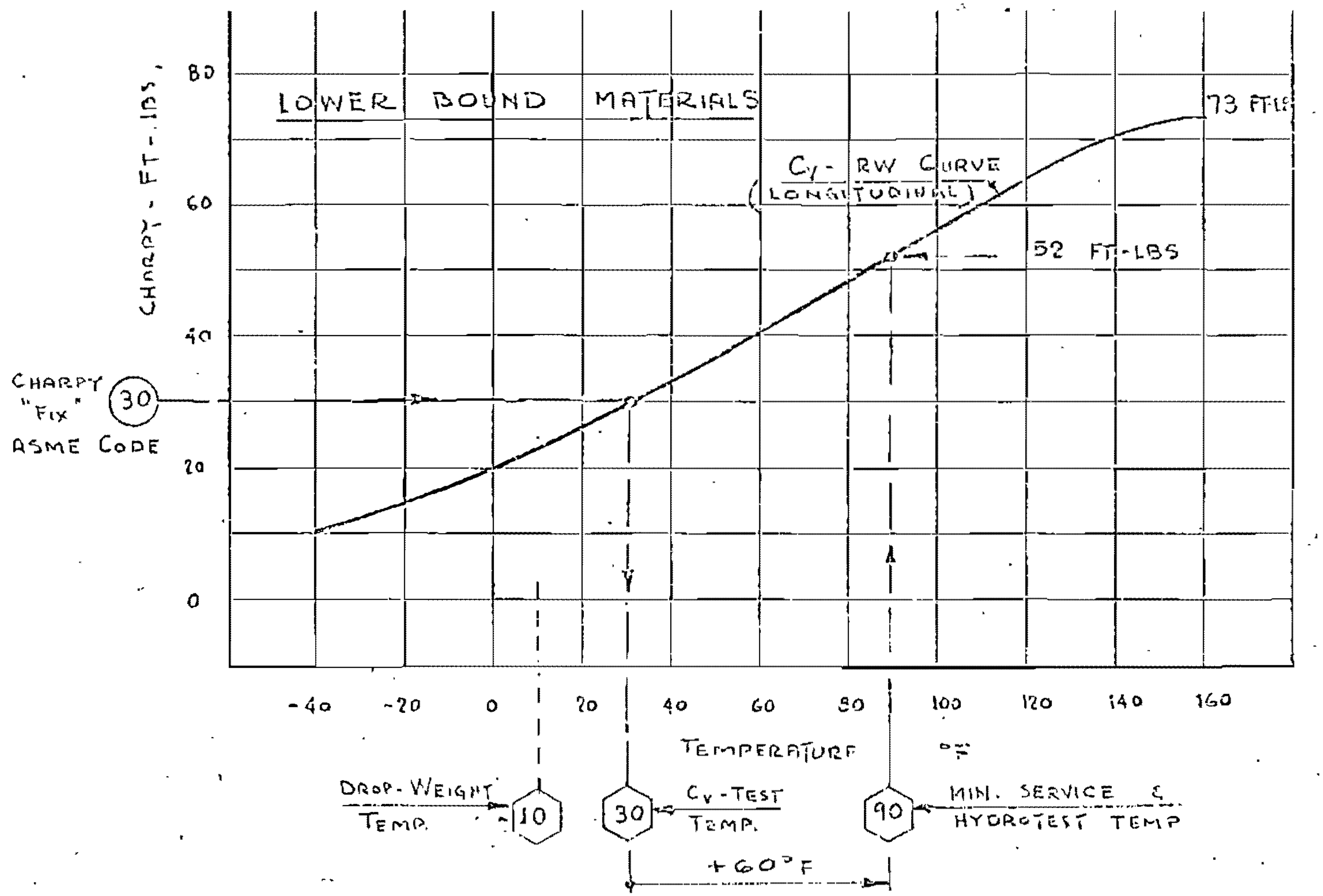
3-30-70

ASME SECT. III - CODE CRITERIA. -APPLIED TO OTHER THAN BELTLINE MATERIALS



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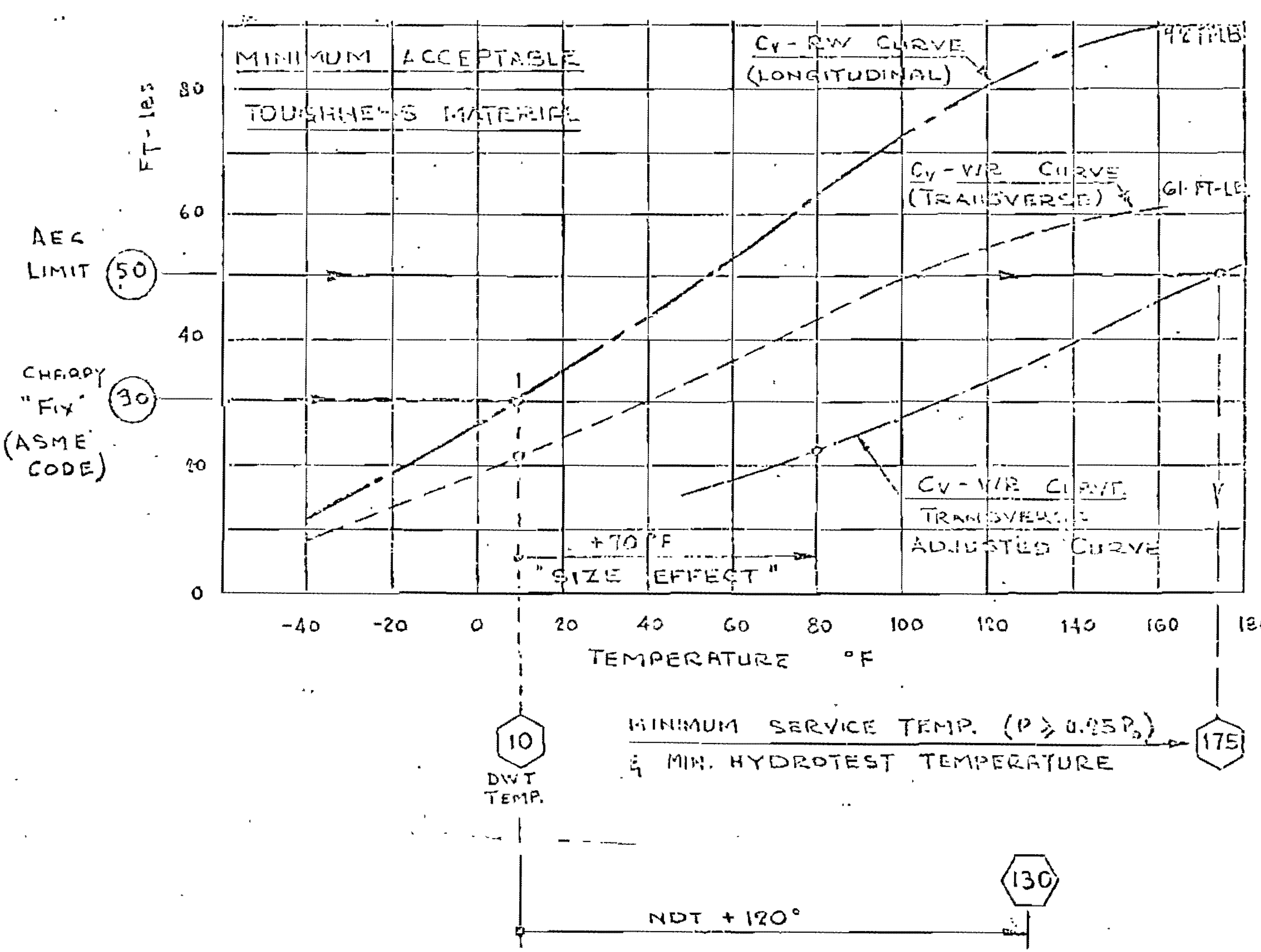
3-30-70

AEC FRACTURE TOUGHNESS CRITERIA APPLIED TO OTHER THAN BELTLINE MATERIALS



USE OF CVN FOR SEMI-DUCTILE OR NON-FRANGIBLE SIMPLE FRACTURE CRITERIA FOR PRESSURE VESSELS

TO OBTAIN A ROUGHLY EQUALLY DUCTILE TYPE FAILURE CRITERIA, FRACTURE MECHANICS AND DROP WT. TESTING GROUPS AGREE

$$\left(\frac{K_{IC}}{\sigma_{YP}}\right)^2 = A \cdot t$$

IS A REASONABLE CRITERIA WHERE:

(1) "A" IS CONSTANT WHICH SHOULD BE LESS THAN $\frac{1}{2.5}$ TO PREVENT PLAIN STRAIN (FOR PURPOSES HERE CHOOSE 0.5 AS AN ARBITRARY NUMBER FOR SAMPLE CALCULATIONS.

(2) "t" IS VESSEL WALL THICKNESS

RELATING CHARPY ENERGY TO ~~[illegible]~~ ~~[illegible]~~ THIS FRACTURE DUCTILITY CRITERION ~~[illegible]~~ FOLLOWS FROM BARSOM'S RELATIONSHIP

$$\left(\frac{K_{IC}}{\sigma_{YP}}\right)^2 = \frac{5}{\sigma_{YP}}\left[CVN - \frac{\sigma_{YP}}{20}\right]$$

WHICH APPEARS TO BE A COMPREHENSIVE RELATIONSHIP FOR ALL STEELS WHERE: {CVN IS CHARPY UPPER SHELF}

EQUATING THESE RELATIONSHIPS

$$CVN = \sigma_{yp}\left[\frac{At}{5} - \frac{1}{4}\right]$$

THIS RELATIONSHIP CANNOT BE REGARD AS A DETAILED CRITERIA, BUT IT IS CERTAINLY LESS CRUDE THAN USING A SINGLE CHARPY NUMBER FOR ALL YIELD STRENGTHS AND VESSEL THICKNESSES. LET US COMPARE CURRENT EXTREME CONDITIONS IN ORDER TO OBSERVE WHETHER IT SHOWS SIGNIFICANT TRENDS.

ASSUME:

(1) A = 0.5 FOR FAIRLY FRANGIBLE BEHAVIOR

(2) VESSEL THICKNESS OF 6" AND 12" (CURRENT DESIGN EXTREMES)

(3) UNIRRADIATED TO IRRADIATED MATERIAL HAVING A YIELD STRENGTH FROM 50 ksi TO 100 ksi

UNDER THESE CONDITIONS THE CVN REQUIRED WOULD SHOW:

FOR 6" THICK, $\sigma_{yp} = 50$ ksi (UNIRRADIATED)

$$CVN = 50\left[\frac{6''}{10} - \frac{1}{4}\right] = 17.5 \text{ FT LBS}$$

.60
.25
.35

FOR ~~10~~ 6" THICK, $\sigma_{yp} = 100$ ksi (HEAVILY IRRADIATED)

$$CVN = 100\left[\frac{6''}{10} - \frac{1}{4}\right] = 35 \text{ FT. LBS.}$$

1.20
.25
.95

FOR 12" THICK, $\sigma_{yp} = 50$ ksi (UNIRRADIATED)

$$CVN = 50\left(\frac{12''}{10} - \frac{1}{4}\right) = 47.5 \text{ FT. LBS.}$$

FOR 12" THICK, $\sigma_{yp} = 100$ ksi (HEAVILY IRRADIATED)

$$CVN = 100\left(\frac{12''}{10} - \frac{1}{4}\right) = 95 \text{ FT. LBS.}$$

NOW THE PROPOSED 50 FT-LB LEVEL AS A SINGLE NUMBER OBVIOUSLY DOES NOT REFLECT THE STRONG ROLL OF THICKNESS IN RELATIVE SAFETY. ADMITTEDLY THE THICK VESSEL MIGHT NOT BE RELATIVELY

AS BAD AS REFLECTED HERE SINCE:

(1) IRRADIATION DAMAGE WOULD NOT BE UNIFORM THRU THE WALL. (BOTH TOUGHNESS AND YIELD STRENGTH VARIATIONS)

(2) PERHAPS IN A THICKER VESSEL WITH CAREFULL INSPECTION PRACTICE, FLAW SIZES MIGHT BE A SMALLER RELATIVE SIZE IN PROPORTION TO THICKNESS.

(3) OPERATING TRANSIENT CONDITIONS MIGHT BE LESS SEVERE.

HOWEVER, NONE OF THESE FACTORS OR OTHERS ELIMINATE THE STILL RATHER STRONG ROLL OF VESSEL THICKNESS IN ASSESSING FRACTURE POSSIBILITIES. MOREOVER FOR EQUAL FRACTURE SAFETY THE CVN SHOULD BE TIED DIRECTLY TO THE YIELD POINT OF THE MATERIAL. THUS LOOKING AT THE CALCULATED REQUIRED CVN NUMBERS ABOVE SHOULD BE VIEWED AS RATHER ROUGH TRENDS FOR SPECIFYING EQUAL FRACTURE SAFETY. THE TRENDS APPEAR TO BE STRONG ENOUGH AS TO BE APPROPRIATE FOR SOME CONSIDERATION

IN SPECIFYING FRACTURE SAFETY.

CONCLUSIONS

(1) THE IDEA OF SPECIFYING A MINIMUM CHARPY UPPER SHELF AND A MINIMUM OPERATING TEMPERATURE FOR A REACTOR IS A RATHER REASONABLE SUGGESTION.

(2) USING THE SAME CVN NUMBER AS THE CRITERIA FOR ALL VARIATIONS IN YIELD STRENGTH AND REACTOR WALL THICKNESSES MAY BE POSSIBLE WITH A VERY CONSERVATIVE CRITERIA. BUT THIS POINT SHOULD BE VERY CAREFULLY REVIEWED WITH RESPECT TO WORST CONDITIONS (THICK REACTORS, HEAVILY IRRADIATED, MINIMUM ACCEPTABLE OPERATING PRACTICE, ETC.)

(3) USING A SINGLE CVN NUMBER CRITERIA WILL TEND TO NOT PENALIZE THE DESIGNER FOR A LESS SAFE DESIGN, WHICH IS TO ME A DOUBTFUL WAY TO FORMULATE SPECIFICATION PHILOSOPHY.

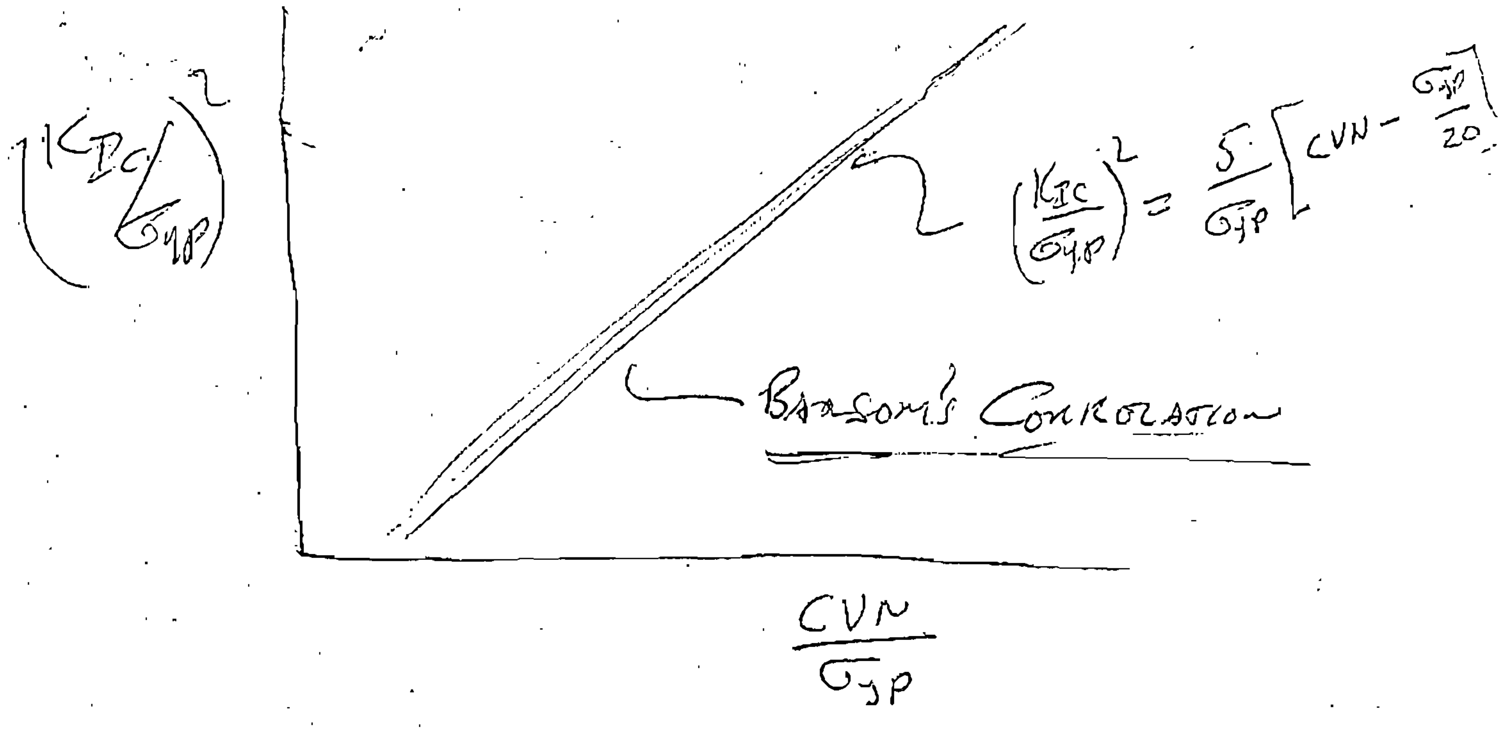
(4) IT SEEMS SENSIBLE THAT IF A VERY CONSERVATIVE CVN TYPE SPECIFICATION IS MADE THAT THE SPECIFICATION SHOULD ALSO ALLOW RESORTING TO MORE PRECISE FRACTURE MECHANICS CALCULATIONS

~~[illegible]~~ TO PROVE SAFETY IF THE CVN SPECIFICATION ~~[illegible]~~ CANNOT QUITE BE MET.

[(5) I AM PERSONALLY STILL MOST WORRIED ABOUT THE SAFETY OF THE THICKER - HEAVILY IRRADIATED REATORS.]

[signature]
4/3/70

FOR ACRS COMMITTEE WORK ONLY (i.e COMMITTEE, CONSULTANTS, RELATED AEC STAFF DISTRIBUTION ONLY).



TO ASSURE NON-FRANGIBLE BEHAVIOR $\Big\} \rightarrow \left(\frac{K_{IC}}{\sigma_{yp}}\right)^2 = A\,t^{(0.5?)}$ (FOR σ_{DESIGN} = GIVEN % OF σ_{yp})

t = WALL THICKNESS $\rightarrow$ A = CONSTANT (DEGREE OF SHEAR)

$$\left(\frac{K_{IC}}{\sigma_{yp}}\right)^2 = \frac{5}{\sigma_{yp}}\left[CVN - \frac{\sigma_{yp}}{20}\right]$$

EQUATING FOR EQUAL SAFETY

$$\frac{5}{\sigma_{yp}}\left[CVN - \frac{\sigma_{yp}}{20}\right] = A\,t^{(0.5)}$$

$$CVN = \frac{A\,t\,\sigma_{yp}}{5} + \frac{\sigma_{yp}}{20}$$

$$CVN = \sigma_{yp}\left[\frac{At}{5} + \frac{1}{20}\right]$$

REQUIREMENT "FORM"

[signature]
4/3/70

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DRAFT
February 19, 1970
Revised March 18, 1970

10 CFR Part 50

Contents

§ 50.55a (i) FRACTURE TOUGHNESS CRITERIA

(1) Introduction

These criteria specify minimum fracture toughness requirements for ferritic materials used in pressure-containing components of the reactor coolant pressure boundary to provide an adequate margin of safety against ~~brittle~~ fracture over the entire service lifetime under the system design conditions of the reactor coolant pressure boundary.

For the purpose of these criteria, the system service conditions include those pressure and temperature transients imposed by normal reactor operation, system hydrostatic tests, and other loading transients specified in the respective design specification of the pressure containing components.

Include comment that the applicants are required to demonstrate a new approach is satisfactory tho not covered by this criterion.

(2) Definitions

(i) "Pressure-~~containing~~ -retaining components" means those pressure vessels, piping, valves, and pumps, including pressure-retaining bolting thereof, which make up the reactor coolant pressure boundary.

(ii) "Reactor coolant pressure boundary" is defined in 10 CFR Part 50, § 50.2.

(iii) "Ferritic materials" means those carbon and low alloy steels with specified minimum tensile strength less than 100,000 psi, (including welds and weld heat-affected zones in such materials), and low alloy steel bolting with specified minimum tensile strength not greater than 125,000 psi, which conform with the specifications as identified in the applicable construction code under which rules the component is built.

POSSIBLY DELETE

ADD UPPER LIMIT TO TENSILE STR.

RELOCATE

ADD WORDS LIMITING THIS CRITERION TO THESE NON-IRRADIATED MATERIALS

(iv) "Normal reactor operation" ~~are~~ those service conditions normally expected during operation of the reactor coolant system, including, but not limited to:

Normal system startup

Normal operation in the design power range (includes hot standby)

System transients in changing from one normal condition to another (e.g., power loading and unloading)

LIST THEM, DECIDE WHAT'S NORMAL, EMERGENCY, ETC.

(v) "System hydrostatic tests" includes those pressurization cycles to which the reactor coolant pressure boundary, or portions thereof, will be subjected in the conduct of any hydrostatic tests. Such tests include those required to comply with the rules of the ASME Section XI - "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems" as well as tests conducted prior to each plant startup.

FIND BETTER WORD.

(vi) "Minimum service temperature" of a component is the minimum temperature above which the coincident pressure imposed on the component may exceed 25 percent of the reactor coolant system operating pressure (at normal rated power), or 25 percent of the reactor coolant system hydrostatic test pressure.

ADD COMMENT ON SUITABLE OPER. RESTRICTIONS

(vii) "Adjusted fracture energy", for the purpose of these criteria, shall be the fracture energy of ferritic material, at a given temperature, ~~obtained from the Charpy V-notch curve adjusted~~ in accordance with (5)(ii)(a).

(3) Fracture Toughness Requirements - Outset of Service-Life

(i) Ferritic materials of pressure-containing components of the reactor coolant pressure boundary shall exhibit, at the outset of service-life, the following fracture toughness properties AT THE MIN. SERVICE TEMP. IN 2 (vi) under the pressure and temperature transients of the system service conditions associated with normal reactor operation, and system hydrostatic tests;

(ii) ~~Ferritic materials~~ (except as qualified under (3)(v)) shall exhibit, ~~at the minimum service temperature~~, adjusted fracture energy levels no lower than:

(a) For plates, pipes, tubes, forgings, and castings:

ADD <12 FOR VESSEL ~~WALLS~~ SHELL PLATES OR FORGINGS

Section Thickness t (inches)		Minimum Charpy-V-Notch Adjusted Fracture Energy (ft-lbs)
t ≥ 3	5 → DRS to study	50
1 < t < 3	2.5	40
t ≤ 1	2	~~30~~ 35

STAFF TO LOOK AT EXPERIENCE.

(b) For pressure-retaining bolting: SEPARATE CONSIDERATION

Diameter d (inches)	Charpy-V-Notch Adjusted Fracture Energy (ft-lbs)
d ≥ 3	50
d < 3	40

(iii) The upper shelf fracture energy levels, as determined by Charpy V-notch tests, shall be at least ~~10~~ 15 (but to be checked) ft-lb higher than the values specified under (3)(ii), except for reactor vessel beltline material which shall meet additional requirements of (3)(iv).

(iv) Reactor vessel beltline material, whose fracture toughness properties may be subject to significant degradation in service, shall exhibit upper shelf energy levels significantly higher than those required under (3)(iii). The adequacy of such initial upper shelf energy levels shall be justified by appropriate data and analyses based on the estimated service degradation of fracture toughness properties. Such analyses shall be made available for review by the Commission.

(v) Ferritic material 1/2-inch and less in thickness, when made to fine-grain practice, may be used in pressure containing components of the reactor coolant pressure boundary without compliance with the requirements of (3)(ii) provided their minimum service temperature is not less than 100° F.

(4) Inservice Requirement - Reactor Vessel Beltline Material

(i) Reactor vessels constructed of ferritic materials shall have their beltline region materials and weld properties monitored by a surveillance program conforming to the requirements of "Reactor Material Surveillance Program".

(ii) Reactor vessels shall be designed to permit the conduct of a thermal annealing treatment to recover material toughness properties of the reactor vessel beltline ferritic materials, except unless experimental data and tests performed on comparable reactor vessel steels demonstrate that the adjusted fracture energy of the reactor vessel beltline material will meet conservatively the requirements of (3)(ii) at a temperature of 200°F over the entire service lifetime of the reactor vessel.

MAKE STRONGER

(iii) Reactor vessels may be permitted continued operation provided the adjusted fracture energy, at the minimum service temperature, as established at any service period from the test results of the material surveillance program satisfies the limit specified in (3)(ii).

WORK IN FLAWS.

(iv) In the event the adjusted fracture energy, at the minimum service temperature, as established at any service period from the test results of the material surveillance program, falls between 50ft-lbs, and 35-ft-lbs, the reactor vessel may be permitted continued operation provided:

REVIEW

ADD REFERENCE TO THICKNESS.

(a) Adequate safety margins for continued operation can be demonstrated by the applications of the principles of fracture mechanics based on valid fracture toughness data obtained from representative materials including welds, and weld heat-affected zones. REFER TO ASTM SPECS AND K_{IC}

(b) Essentially 100 percent VOLUMETRIC inservice inspection of the beltline region of the reactor vessel is performed to establish the existence, if any, of significant flaws in the material.

(c) A fracture mechanics analysis is performed based on: the beltline stress analyses, the fracture toughness properties of the materials obtained from (4)(iv)(a), and the inservice inspection of (4)(iv)(b).

Such test data and analyses shall be made available for review and approval by the Commission.

(v) In the event the adjusted fracture energy, at the minimum service temperature, as established at any service period from the test results of the material surveillance program, falls below 35ft-lbs, the reactor vessel may no longer be permitted continued operation, unless a thermal annealing treatment is performed to effect a recovery of material toughness properties of the reactor vessel beltline material.

The proposed annealing method and procedure, and the fracture toughness test data before and after annealing treatment, shall be made available for review and approval by the Commission.

(5) Fracture Toughness Tests

(i) Ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch impact test (ASTM A-370), and the Dropweight test (ASTM E-208), in accordance with the following requirement and the adjusted fracture energy levels determined as specified in (5)(ii) to demonstrate compliance with the fracture toughness criteria of (3)(ii):

(a) Charpy V-notch (C_v) impact tests shall be conducted to define the C_v test curve (including the upper-shelf energy level) using Type A specimens oriented with respect to the "weak" direction (WR orientation in plates) of the respective plates, forgings, castings, pipe, and tubes intended for pressure-containing components.

(b) Where the use of specimens oriented with respect to the "weak" direction is not practicable, specimens oriented with respect to the "strong" direction (RW orientation in plates) may be used provided test correlation data obtained from ferritic materials of the same specification is available to convert the C_v test curve (RW orientation) to the C_v test curve (WR orientation). Where such correlation is not available, the "strong" direction C_v test curves may be used to demonstrate adequate fracture toughness provided that ferritic materials exhibit, at the minimum service temperature, adjusted fracture energy levels no lower than 1.5 times the energy levels of (3)(ii).

LOOK AT ROLLING RATIOS

(c) Materials used to prepare test specimens shall be representative of the actual properties of the finished component as required by the applicable construction code under which rules the component is built, with the exception that ferritic materials intended for the reactor vessel beltline region shall comply with the requirements of (5)(i)(d).

(d) 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 WHICH HAVE GONE THROUGH THE WHOLE PROCESS. Where seamless shell forgings are used, the test specimens shall be taken from a separate weldment using excess material from the shell forging(s) and welded under the same production, welding conditions applied in joining the shell forgings.

(e) Charpy V-notch impact test machines used to determine fracture toughness properties for comparison with the criteria of (3)(ii) shall have been calibrated at least once in each 6-month interval employing standard specimens obtained from U. S. Army Materials Research Center.
REFER TO ASTM SPEC.

(f) Temperature instrumentation used to control test temperature of specimens, for both Charpy V-notch impact tests and drop-weight tests shall have been calibrated at least once in each 3-month interval.

(ii) Adjusted Fracture Energy

The Charpy V-notch (C_v) test curve as derived from the tests of (5)(i) shall be adjusted as follows, to establish the adjusted fracture energy of each material tested, for comparison with the acceptance requirements specified in (3)(ii):

(a) The Charpy V-notch curve of (5)(i) shall be translated to the right along the temperature coordinate by a temperature increment equal to the sum of:

(1) the difference between the Nil-Ductility Transition (NDT) temperature derived from the Dropweight test (DWT), and the temperature corresponding to the Charpy V-notch "fix" energy value (ft-lbs) specified in the ASME Section III - Nuclear Vessel Code for the applicable ferritic material, to be applied only when the NDT temperature is higher than the temperature corresponding to the Charpy V-notch "fix," and

REVIEW

(2) a "size-effect" increment of 70°F to be applied only for material thickness above 3-inches,

(b) The adjusted fracture energy, as read from the adjusted C_v curve of (5)(ii)(a), at the minimum service temperature, shall be used to determine compliance with the fracture toughness requirement of (3)(ii).

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4/21/70

DRAFT
February 20, 1970

Appendix F

Reactor Material Surveillance Program Requirements

Contents

APPENDIX F

REACTOR MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I Introduction

The requirements of this material surveillance program are intended to monitor the changes in the fracture toughness properties of reactor vessel beltline region constructed of ferritic materials, which changes such materials may experience as a consequence of neutron irradiation over the service life of the vessel. The purpose of the program is to obtain fracture toughness test data from material specimens withdrawn periodically from the reactor vessel, to assure that the vessel will be operated under conditions of adequate margins of safety against ~~brittle~~ fracture.

II Definitions

A. "Beltline region of reactor vessels" means the shell material, including welds and weld heat-affected zones, which directly surround the effective height of the reactor core.

REWORD TO ACCOMODATE AREAS ABOVE + BELOW CORE

B. "Effective height of reactor core" is not less than the overall height of the reactor fuel element assemblies, and in no case less than the height of vessel internal thermal shields where used.

C. "Minimum service temperature" is defined in 10 CFR Part 50, § 50.55a (i)(2).

D. "Adjusted fracture energy" is defined in 10 CFR Part 50, § 50.55a (i)(2).

E. "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 <u>Material Surveillance Program</u>

A. Surveillance Program Requirements - Reactor vessels constructed of ferritic materials shall have their beltline region monitored by a surveillance program complying with the following requirements and the provisions of the ASTM Specification E-185-70 except as modified by these requirements:

1. Surveillance specimens shall be taken directly from the excess shell course material, welds, and heat-affected zones of the beltline region of the reactor vessel, which are used to conduct the fracture toughness tests in meeting the requirements of 10 CFR Part 50, § 50.55a (i)(4).

2. Irradiation capsules containing the surveillance specimens shall be located as close as practical ~~practicable~~ to the inside vessel wall. In any case, the capsule locations shall be such that the calculated neutron flux received by the innermost (with respect to the reactor core) irradiation specimens will not exceed three times the calculated maximum neutron flux at the inside wall of the vessel. The design and location of the capsules shall permit removal and reinsertion of the capsules.

3. The required number of specimens and capsules and their withdrawal schedule shall be governed by the following:

(a) For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted fracture energy level of the reactor vessel beltline region will meet the requirements of § 50.55a (i)(3)(ii), at a temperature of 100°F, over the entire service lifetime of the reactor vessel, at least three capsules shall be required, for withdrawal as follows:

Withdrawal Schedule

1st capsule	1/4 Service Life
2nd capsule	3/4 Service Life
3rd capsule	Standby

In the event, the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the Charpy V-notch (C_v) fracture energy curve greater than orginally predicted by test data, the withdrawal schedule shall be modified as follows:

Withdrawal Schedule

1st capsule	1/4 Service Life
2nd capsule	1/2 Service Life
3rd capsule	Standby

INCLUDE MONITOR REQUIREMENT

(b) For reactor vessels which do not meet the conditions of III A.3(a) but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted fracture energy levels of the reactor vessel beltline region will meet the requirements of § 50.55a (i)(3)(ii), at a temperature of 200°F over the entire service lifetime of the reactor vessel, at least four capsules shall be required, for withdrawal as follows:

Withdrawal Schedule

1st capsule	At time when predicted shift of C_v adjusted fracture energy curve is approx. 50°F or 1/4 of service life, whichever comes first.
2nd capsule	Approx. equal interval between 1st and 3rd capsule withdrawal
3rd capsule	3/4 Service Life
4th capsule	Standby

(c) For reactor vessels which do not meet the conditions of III A.3(b), at least five capsules shall be required, for withdrawal as follows:

Withdrawal Schedule

1st capsule	At time when predicted shift of C_v adjusted fracture energy curve is approx. 50°F or 1/4 etc.
2nd & 3rd capsule	Approx. equal intervals between 1st and 4th capsule
4th capsule	3/4 of Service Life
5th capsule	Standby

(d) Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

(e) Sufficient "archive" material shall be retained to prepare additional surveillance specimens (as recommended by ASTM Specification E-185-70) except for reactors which meet the conditions of III A.3(a) or (b). The "archive" material shall be obtained from the excess shell course material, welds, and heat affected zones as identified in III A.1.

B. Integrated Surveillance Program

1. An integrated surveillance program may be employed for multiple reactor vessels located at one site, provided that:

(a) The reactors are designed in accordance with the same design specification, and constructed by a single fabricator using the materials produced to the same specifications, and employing the same fabrication procedures.

design and

(b) All reactor vessels meet the conditions of III A.3(a) or (b).

CHANGE CONCEPT TO INCLUDE SOME CAPSULES FOR ALL REACTORS

(c) All reactors will be operated under comparable conditions and service.

(d) All reactors are provided with equivalent space for insertion of a full complement of surveillance capsules and sufficient "archive" material is retained for this purpose.

(e) Each vessel contains material specimens obtained from its respective beltline region as required by the provisions of III A.1.

(f) The most conservative value of adjusted fracture energy levels determined from tests of the specimens withdrawn from any of the reactors will be applied to all vessels in establishing operational limitations.

2. For an integrated surveillance program, the required number of capsules and the withdrawal schedules shall conform to the following:

REWORD

(a) For reactor vessels which meet the conditions of III A.3(a), the number of capsules and the withdrawal schedule shall conform to the requirements of III A.3(a).

(b) For reactor vessels which do not meet the conditions of III A.3(a), the vessel initially placed in service shall contain a full complement of capsules as required by III a.3(b) ~~or (c), as applicable~~. The remaining vessels shall contain a minimum of three capsules.

The withdrawal schedule for the vessel with the full complement of capsules shall comply with the schedule specified in III A.3(b) ~~or (c), as applicable~~.

The withdrawal schedule for the other vessels shall correspond approximately to the schedule for the withdrawal of the last two capsules from the vessel initially placed in service, and the third capsule retained as a standby.

C. Fracture Toughness Tests

1. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of 10 CFR Part 50, § 50.55a (i)(5) "Fracture Toughness Tests."

2. The test results shall be adjusted in accordance with the procedure specified under 10 CFR Part 50, § 50.55a (i)(5) to verify that the fracture toughness requirements of § 50.55a (i)(3)(ii) are satisfied.

D. Report of Test Results

1. Each specimen withdrawal and the fracture toughness tests shall be the subject of a summary technical report, which includes a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the translation of the measured results to those expected in the reactor vessel beltline region.

   The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the orginally predicted values.

   The minimum service temperature, established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes in operational procedures which will be adopted to assure meeting such temperature limitations.

return to → ~~B. Grimes~~ ~~D. Knuth~~ ~~A. Tedesco~~ ~~[illegible]~~

bcc wo/encls:
HLPrice, DR
RRMaccary, DRS
SPawlicki, DRS

bcc w/encls:
CKBeck, DR
EDCase, DRS
PAMorris, DRL
T. Schroeder, DRL
RSBoyd, DRL
RCDeYoung, DRL
DJSkovholt, DRL
RSBrodsky, NR
MBooth, RDT
RGSmith, DRS

APR 23 1970

Dr. Joseph M. Hendrie, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D.C. 20545

Dear Dr. Hendrie:

At the April 1, 1970, ACRS Subcommittee meeting, the following draft criteria concerning the fracture toughness and surveillance program requirements for ferritic materials used in the reactor coolant pressure boundary were reviewed:

10 CFR Part 50 § 50.55(d) - Fracture Toughness Criteria, Draft, February 19, 1970

Appendix F - Reactor Material Surveillance Program Requirements, Draft, February 20, 1970

These criteria have been revised to reflect suggestions and comments from ACRS members at the meeting and are now dated April 17 and April 7, 1970, respectively.

Enclosed are eighteen copies of these revised documents. ACRS review and comment are requested as soon as possible. We believe that these criteria, in their present form, are suitable for publication for comments in the Federal Register.

Sincerely,

[illegible]

Edson G. Case, Director
Division of Reactor Standards

Enclosures:
As stated

OIM-7 ACRS

OFFICE	DRS:MEB	DRS:ADIR	DRS:DIR			
SURNAME	Pawlicki:jjb	Maccary	Case			
DATE	4/21/70	4/?/70	4/?/70			

DRAFT
April 17, 1970

10 CFR Part 50

Contents

§ 50.55a (1) FRACTURE TOUGHNESS CRITERIA

(1) Introduction and Scope

These criteria specify minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, to provide adequate margins of safety against fracture under normal reactor operation, system hydrostatic tests, and anticipated transients to which the system may be subjected over its entire service lifetime.

These criteria apply to carbon and low alloy ferritic steels (including welds and weld heat-affected zones in such materials) whose specified minimum yield strength, as defined in (2)(v), does not exceed 50,000 psi. Adequacy of fracture toughness of materials with higher specified minimum yield strength shall be subject to review and approval by the Commission.

These criteria apply to pressure-retaining components of the reactor coolant pressure boundary for nuclear power plants whose construction permit is issued   months after these criteria are published as a Regulation. For nuclear power plants whose construction permit is issued prior to that date, these criteria shall be applied to the extent practicable, to establish safe operating procedures based on the measured fracture toughness of the ferritic materials used. In cases where these criteria cannot be satisfied at any time during the service life, the licensee shall demonstrate by other means that adequate margins of safety are available for continued operation of the system.

The minimum fracture toughness requirements specified in these criteria are based on the current state of development of fracture mechanics and the available fracture toughness data. These criteria are subject to periodic revisions as more data are obtained on the fracture toughness properities of irradiated heavy section steels.

(2) Definitions

(i) "Pressure-retaining components" are those pressure vessels, piping, valves, and pumps, which make up the reactor coolant pressure boundary, as defined in 10 CFR Part 50, § 50.2.

(ii) "Normal reactor operation" includes those conditions normally expected during the course of reactor coolant system operation, including, but not necessarily limited to:

(a) normal system startup

(b) normal operation in the design power range (including hot standby operation)

(c) normal system shutdown and cooldown

(d) system transients in changing from one normal condition to another (e.g., power loading and unloading)

(iii) "System hydrostatic tests" include those pressurization cycles to which the reactor coolant pressure boundary, or portions thereof, will be subjected in the conduct of any hydrostatic tests of the system. Such tests include those required to comply with the rules of the ASME Section XI - "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems" as well as tests conducted prior to each plant startup.

(iv) "Anticipated transients" include faults of moderate frequency, and infrequent faults. Faults of moderate frequency are those deviations from normal operating conditions under which the reactor coolant system must regain its normal operational status, including, but not necessarily limited to:

(a) single errors by operators in the use of controls or control devices, which under normal operating conditions are corrected automatically,

(b) reactivity changes due to improper control rod motions, chemical (neutron absorber) dilutions, or inadvertent moderator cooldown,

(c) step-load transients including reactor-turbine load mismatch (e.g., loss of load),

(d) reactor scrams during normal reactor operating conditions (e.g., turbine trip),

(e) reactor coolant flow interruptions (e.g., pump failure, loss of pump power),

(f) malfunctions of the associated auxiliary systems, and the main steam and feedwater systems, which cause transients in the reactor coolant system.

Infrequent faults are those postulated deviations from normal operating conditions which require the reactor to be shut down, including, but not necessarily limited to:

(g) system depressurization by failure of active elements, (e.g., failure of a safety valve to open, or inadvertent opening of a normally closed valve),

(h) reactivity excursion resulting from the ejection of maximum worth control rod,

(i) transients in reactor coolant system as a result of a major rupture in the boundary of the main steam and feedwater systems,

(j) design postulated accidents (e.g., rupture of any pipe within the boundary of the reactor coolant system) which require operation of emergency core cooling systems,

(k) reactor coolant systems transients produced by the operation of emergency core cooling systems.

(v) "Specified minimum yield strength" is the minimum yield strength in the unirradiated condition of a material, tabulated in the construction code under which rules the component is built.

(vi) "Lowest pressurization temperature" of a component is the lowest temperature at which coolant pressure within the component exceeds 25 percent of the reactor coolant system operating pressure, or at which the rate of temperature change in the component material exceeds 50°F/hr, under normal reactor operation, system hydrostatic tests or anticipated transients.

(viii) "Adjusted fracture energy" is the fracture energy of ferritic material, at a given temperature, obtained from the Charpy V-notch curve adjusted in accordance with (5)(ii)(a).

(viii) "Beltline region of reactor vessel" comprises the shell material, including welds and weld heat-affected zones, which directly surrounds the effective height of the fuel element assemblies, plus any additional material for which the predicted shift of the Charpy V-notch (C_v) fracture energy curve exceeds 100°F.

(3) Fracture Toughness Requirements

(i) Ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (except as qualified under (3)(v)) shall exhibit throughout their service lifetime, at the lowest pressurization temperature, adjusted fracture energy levels no lower than the following.

Section Thickness t (inches)	Minimum Charpy-V-Notch Adjusted Fracture Energy (ft-lbs)	
	Average 3 specimens	One Individual Specimen
[illegible]	[illegible]	45*
[illegible]	45	40
t [illegible] 2	40	35

(ii) The upper shelf fracture energy levels, as determined by Charpy V-notch tests, shall be at least 15ft-lb higher than the values specified under (3)(i), except for reactor vessel beltline material which shall meet additional requirements of (3)(iii).

(iii) [illegible] on the upper shelf fracture energy level [illegible] as determined by

*For reactor vessels [illegible] these minimum fracture energy levels may be inadequate for plates and forgings thicker than 12 inches. The proposed minimum fracture toughness for such vessels shall be subject to a separate review by the Commission.

(3) Fracture Toughness Requirements

(i) Ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (except as quantified under (3)(v)) shall exhibit throughout their service lifetime, at the lowest pressurization temperature, adjusted fracture energy levels no lower than the following:

Section Thickness t (inches)	Minimum Charpy V-Notch Adjusted Fracture Energy (ft-lbs)	
	Average 3 Specimens	One Individual Specimen
t ≥5	50	45*
2 < t < 5	45	40
t ≤ 2	40	35

(ii) The upper shelf fracture levels, as determined by Charpy V=notch tests, shall be at least 15 ft-lb higher than the values specified under (3)(i), except for reactor vessel beltline material which shall meet the additional requirements of (3)(iii).

(iii) For the reactor vessel beltline region the upper shelf fracture energy levels for unirradiated material, as determined by

* For reactor vessel beltline region these minimum fracture energy levels may be inadequate for plates and forgings thicker than 12 inches. The minimum fracture toughness of such vessels shall be subject to a separate review by the Commission.

Charpy V-notch tests, shall meet the following requirements, except where it can be conservatively demonstrated by appropriate data and analyses that lower values of upper shelf fracture energy are adequate. Such data and analyses shall be subject to review and approval by the Commission.

(a) For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels, and making proper allowances for all uncertainties in the measurements, that the adjusted fracture energy level of the reactor vessel beltline region will meet the requirements of (3)(i) at a temperature of 100°F, over the entire service lifetime of the reactor vessel, the upper shelf fracture energy levels for unirradiated material shall meet the requirements of (3)(ii).

(b) For reactor vessels which do not meet the conditions of (3)(iii)(a) but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted fracture energy levels of the reactor vessel beltline region will meet the requirements of (3)(i) at a temperature of 200°F, over the entire service lifetime of the reactor vessel, the upper shelf fracture energy levels for unirradiated material shall be at least 20ft-lbs higher than the values specified under (3)(i).

(c) For reactor vessels which do not meet the conditions of (3)(iii)(b), the upper shelf fracture energy levels for unirradiated material shall be at least 25ft-lbs higher than the values specified under (3)(i).

(iv) Reactor vessels which do not meet the conditions of (3)(iii)(b), shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

(v) Ferritic material 1/2-inch and less in thickness, when made to fine-grain practice, may be used in pressure retaining components of the reactor coolant pressure boundary without compliance with the requirements of (3)(i) provided their lowest pressurization temperature is not less than 100°F.

(vi) The initial fracture toughness properties of unirradiated materials shall be adequate to allow preoperational hydrostatic testing of the reactor coolant pressure boundary, or portions thereof, at temperatures below 200°F.

(4) Inservice Requirement - Reactor Vessel Beltline Material

(i) Reactor vessels shall have their beltline region materials and weld properties monitored by a surveillance program conforming to the "Reactor Material Surveillance Program Requirements", Appendix F.

(ii) Reactor vessels shall be acceptable for continued operation for that service period within which

(a) the predicted adjusted fracture energy, at the lowest pressurization temperature, (as derived from the test results of the material surveillance program of (i) above) satisfies the requirements of (3)(i), and

(b) the requirements of the ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems", are met prior to each resumption of system operation.

(iii) In the event that the requirements of (3)(i) cannot be met, reactor vessels may be acceptable for continued operation, provided the following requirements are satisfied for the specified conditions:

(a) if the predicted adjusted fracture energy level is equal or higher than 35ft-lbs, the beltline region of the vessel shall be subjected to essentially 100 percent volumetric examination in accord with the rules of ASME Code Section XI, and a fracture mechanics analysis shall be performed, which conservatively demonstrates that adequate safety margins exist for continued operation. Such analysis shall be based on:

(1) flaw sizes detected by the inservice inspection,

(2) valid fracture toughness data (K_{IC} per ASTM Spec. E-24) for the irradiated base metal, welds and weld heat-affected zones, and

(3) stress analyses of the beltline region.

(b) if the predicted adjusted fracture energy level is lower than 35ft-lbs, 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 monitored by testing specimens from the surveillance program capsules before and after annealing treatment, and shall be adequate to satisfy the requirements of (3)(i) at the end of the proposed service period.

(c) if the requirements for conditions (a) or (b) cannot be satisfied, the licensee shall be responsible to demonstrate, by other appropriate means, that adequate safety margins exist for continued operation.

The proposed programs for conditions (a), (b) or (c), shall be reported by the licensee to the Commission for review and approval at least 3 years prior to the date when the predicted fracture energy levels will no longer satisfy the reqirements of (3)(i).

(5) <u>Fracture Toughness Tests</u>

(i) Ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch impact test (ASTM A-370), and the unirradiated materials also by means of the Dropweight test (ASTM E-208), in accordance with the following requirement and the adjusted fracture energy levels determined as specified in (5)(ii) to demonstrate compliance with the fracture toughness criteria of (3)(i):

(<u>a</u>) Charpy V-notch (C_v) impact tests shall be conducted to define the C_v test curve (including the upper-shelf energy level) using Type A specimens oriented with respect to the "weak" direction (WR orientation in plates) of plates, forgings, castings, pipe, and tubes intended for pressure-retaining components.

(<u>b</u>) In lieu of (<u>a</u>) above, C_v specimens oriented with respect to the "strong" direction (RW orientation in plates) may be used provided test correlation data obtained from ferritic materials of the same specification are available to convert the C_v test curve (RW orientation) to the C_v test curve (WR orientation).

(<u>c</u>) In lieu of (<u>b</u>) above, C_v specimens oriented with respect to the "strong" direction may be used to demonstrate adequate fracture toughness provided that materials exhibit, at the lowest pressurization temperature, adjusted fracture energy levels no lower than 2 times the energy levels of (3)(i).

(d) Materials used to prepare test specimens shall be representative of the actual properties of the finished component as required by the applicable construction code under which rules the component is built, with the exception that ferritic materials intended for the reactor vessel beltline region shall comply with the requirements of (5)(i)(e).

(e) 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 the heat treatment equivalent to that received by the vessel throughout its fabrication process. Where seamless shell forgings are used, the test specimens shall be taken from a separate weldment using excess material from the shell forging(s) and welded under the same production welding conditions applied in joining the shell forgings.

(f) Charpy V-notch impact test machines used to determine fracture toughness properties for comparison with the criteria of (3)(i) shall have been calibrated at least once in each 6-month interval using methods outlined in ASTM E23-60, and employing standard specimens obtained from U. S. Army Materials Research Center.

(g) Temperature instrumentation used to control test temperature of specimens, for both Charpy V-notch impact tests and dropweight tests shall have been calibrated at least once in each 3-month interval.

(h) Personnel performing these fracture toughness tests shall be qualified by training and experience, and shall have demonstrated compentency to perform the tests in accord with written procedures of the component manufacturer or the licensee.

(i) Fracture toughness tests results shall be recorded and shall include a certification by the responsible person having authority over the tests performed that:

(1) the test data are correctly reported and identified with the material intended for a pressure-retaining component,

(2) the tests have been conducted using machines and instrumentation with available records of periodic calibration, and

(3) the personnel performing the tests are identified and records of their qualifications are available upon request.

(ii) Adjusted Fracture Energy

The Charpy V-notch (C_v) test curve as derived from the tests of (5)(i) shall be adjusted as follows, to establish the adjusted fracture energy of each material tested, for comparison with the acceptance requirements specified in (3)(i):

(a) The Charpy V-notch curve of (5)(i) shall be translated to the right along the temperature coordinate by a temperature increment equal to the sum of:

(1) the difference between the Nil-Ductility Transition (NDT) temperature derived from the Dropweight test (DWT), and the temperature corresponding to the Charpy V-notch "fix" energy value (ft-lbs) specified in the ASME Section III - Nuclear Vessel Code for the applicable ferritic material, as obtained from tests on unirradiated specimens, to be applied only when the NDT temperature is higher than the temperature corresponding to the Charpy V-notch "fix", and

(2) a "size-effect" increment of 80°F to be applied for material thickness 5-inches and greater, and 40°F for material thickness less that 5-inches.

(b) The adjusted fracture energy, as read from the adjusted C_v curve of (5)(ii)(a), at the lowest pressurization temperature, shall be used to determine compliance with the fracture toughness requirement of (3)(i).

DRAFT
April 7, 1970

Appendix F

Reactor Material Surveillance Program Requirements

Contents

APPENDIX F

REACTOR MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I Introduction

The requirements of this material surveillance program are to monitor the changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region as a consequence of neutron irradiation. The purpose of the program is to obtain fracture toughness test data from material specimens withdrawn periodically from the reactor vessel to assure that the vessel will be operated under conditions of adequate margins of safety against fracture throughout its service life.

II Definitions

A. "Beltline region of reactor vessel" is defined in 10 CFR Part 50, § 50.55a (i)(2).

B. "Lowest Pressurization Temperature" is defined in 10 CFR Part 50, § 50.55a (i)(2).

C. "Adjusted fracture energy" is defined in 10 CFR Part 50, § 50.55a (i)(2).

D. "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 Material Surveillance Program

A. Surveillance Program Requirements - Reactor vessels constructed of ferritic materials shall have their beltline region monitored by a surveillance program complying with the provisions of the ASTM Specification E-185-70 except as modified by the following requirements:

1. Surveillance specimens shall be taken directly from the excess shell course material, welds, and heat-affected zones of the beltline region of the reactor vessel, which are used to conduct the fracture toughness tests in meeting the requirements of 10 CFR Part 50, § 50.55a (i)(3). Type of the specimens shall comply with the requirements of § 50.55a (i)(5)(i).

2. Irradiation capsules containing the surveillance specimens shall be located as close as practicable to the inside vessel wall, but shall not be attached to the wall. In any case, the capsule locations shall be such that the calculated neutron flux received by the innermost (with respect to the reactor core) irradiation specimens will not exceed three times the calculated maximum neutron flux at the inside wall of the vessel. The design and location of the capsules shall permit insertion of replacement capsules.

3. The required number of specimens and capsules and their withdrawal schedule shall be governed by the following:

   (a) For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels, and making proper allowances for all uncertainties in the measurements, that the adjusted fracture energy level of the reactor vessel beltline region will meet the requirements of § 50.55a (i)(3)(i), at a temperature of 100°F, over the entire service lifetime of the reactor vessel, at least three capsules shall be required, for withdrawal as follows:

<u>Withdrawal Schedule</u>

1st capsule	1/4 Service Life
2nd capsule	3/4 Service Life
3rd capsule	Standby

In the event the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the Charpy V-notch (C_v) fracture energy curve greater than originally predicted by test data, the withdrawal schedule shall be modified as follows:

3rd capsule	3/4 Service Life
4th capsule	Standby

(c) For reactor vessels which do not meet the conditions of III A.3(b), at least five capsules shall be required, for withdrawal as follows:

<u>Withdrawal Schedule</u>

1st capsule	At time when predicted shift of C_v adjusted fracture energy curve is approx. 50°F or at 1/4 Service Life, whichever comes first.
2nd & 3rd capsule	Approx. equal intervals between 1st and 4th capsule
4th capsule	3/4 of Service Life
5th capsule	Standby

(d) Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

(e) Sufficient "archive" material shall be retained to prepare additional surveillance specimens (as recommended by

ASTM Specificaton E-185-70) except for reactors which meet the conditions of III A.3(a) or (b). The "archive" material shall be obtained from the excess shell course material, welds, and heat-affected zones as identified in III A.1.

B. Integrated Surveillance Program

1. An integrated surveillance program may be employed for multiple reactor vessels located at one site, provided that:

   (a) The reactors are of the same design, ordered to the same design specification, and constructed by a single fabricator using the materials produced to the same specifications, and employing the same fabrication procedures.

   (b) All reactor vessels meet the conditions of III A.3(a) or (b).

   (c) All reactors will be operated under comparable conditions and service.

   (d) Each vessel contains material specimens obtained from its respective beltline region as required by the provisions of III A.1.

(e) The most conservative value of adjusted fracture energy levels determined from tests of the specimens withdrawn from any of the reactors will be applied to all vessels in establishing operational limitations.

2. For an integrated surveillance program, the required number of capsules and the withdrawal schedules shall conform to the following:

(a) For reactor vessels which meet the conditions of III A.3(a), the number of capsules and the withdrawal schedule for each vessel shall conform to the requirements of III A.3(a).

(b) For reactor vessels which meet the conditions of III A.3(b), the number of capsules for each vessel shall conform to the requirements of III A.3(b).

The withdrawal schedule for the vessel initially placed in service shall comply with the schedule specified in III A.3(b).

The withdrawal schedule for the other vessels shall correspond approximately to the schedule for the withdrawal of the last two capsules from the vessel initially placed in service, and the remaining two capsules shall be retained as standbys.

C. Fracture Toughness Tests

1. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of 10 CFR Part 50, § 50.55a (i)(5) "Fracture Toughness Tests."

2. The test results shall be adjusted in accordance with the procedure specified under 10 CFR Part 50, § 50.55a (i)(5) to verify that the fracture toughness requirements of § 50.55a (i)(3)(i) are satisfied.

D. Report of Test Results

1. Each specimen withdrawal and the fracture toughness tests shall be the subject of a summary technical report, which includes a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the translation of the measured results to those expected in the reactor vessel beltline region.

The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values.

The lowest pressurization temperature, as defined in II.B, established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes in operational procedures which will be adopted to assure meeting such temperature limitations.

the Administrator before taking action upon the proposed rule. The proposals contained in this notice may be changed in the light of comments received. All comments will be available, both before and after the closing date for comments, in the Rules Docket for examination by interested persons.

In consideration of the foregoing, it is proposed to amend § 39.13 of Part 39 of the Federal Aviation Regulations by adding the following new airworthiness directive:

GRUMMAN. Applies to all Model G-159 airplanes.

Compliance required as indicated.

To detect cracking in the wing to fuselage attachment fittings at Butt Line 9 of Grumman Model G-159 airplanes, accomplish the following:

a. Within 6 months time in service after the effective date of this AD, unless already accomplished, inspect the wing to fuselage attachment fittings, P/Ns 159WM10064 and 159WM10065 (P/N 159WM10223 assembly), and P/N 159WM10045 at Butt Line 9 Left and Right, wing front beam for cracks, deformation, gaps, or improper shimming in accordance with Grumman Gulfstream I Aircraft Service Change No. 190, dated June 28, 1971, or later FAA approved revision or in a manner approved by the Chief, Engineering and Manufacturing Branch, FAA Southern Region.

b. If cracks, deformation, gaps, or improper shimming are found when conducting the inspection required by paragraph a. within 100 hours time in service after detection correct in accordance with Aircraft Service Change 190 or in a manner approved by the Chief, Engineering and Manufacturing Branch, FAA Southern Region.

c. Upon request of the operator and FAA Maintenance Inspector, subject to prior approval of the Chief, Engineering and Manufacturing Branch, FAA Southern Region, the initial inspection time may be adjusted to coincide with inspections for wing corrosion required by AD 67-4-1.

This amendment is proposed under the authority of sections 313(a), 601, and 603 of the Federal Aviation Act of 1958 (49 U.S.C. 1354(a), 1421, 1423) and of section 6(c) of the Department of Transportation Act (49 U.S.C. 1655(c)).

Issued in East Point, Ga., on June 15, 1971.

JAMES G. ROGERS,
Director, Southern Region.

[FR Doc.71-9454 Filed 7-2-71;8:48 am]

ATOMIC ENERGY COMMISSION

[10 CFR Part 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Fracture Toughness Requirements for Nuclear Power Reactors

The Atomic Energy Commission has under consideration amendments of its regulations in 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add new appendices entitled "Fracture Toughness Requirements" and "Reactor Vessel Material Surveillance Program Requirements." The purpose of the proposed amendments is to specify minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary and to require surveillance of the fracture toughness specimens of the reactor vessel material by periodic tests. These amendments would apply only to boiling and pressurized water power reactors.

Criterion 31 of the "General Design Criteria for Nuclear Power Plants" (Appendix A of Part 50) states that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (a) the boundary behaves in a nonbrittle manner and (b) the probability of rapidly propagating fracture is minimized. The criterion also requires that the design reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (a) material properties, (b) the effects of irradiation on material properties, (c) residual, steady-state and transient stresses, and (d) size of flaws.

The proposed amendments would specify minimum fracture toughness requirements needed to assure that Criterion 31 is satisfied and describe methods by which the fracture toughness of reactor coolant pressure boundary materials should be determined. Because of the special importance to safety of the reactor vessel and because the fracture toughness properties of the reactor vessel beltline region may change as a result of neutron irradiation, special requirements for periodic testing of irradiated specimens of reactor vessel beltline materials would be specified.

Recent fracture toughness test data indicate that the rules of currently applicable industry codes pertaining to the required fracture toughness properties of ferritic materials used in nuclear powerplants may not assure, in some cases, adequate margins of safety under certain conditions of operations. The proposed fracture toughness criteria are based on the theoretical methods of elastic fracture mechanics, currently under further development under the AEC-funded Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory and on recent fracture toughness test data obtained by organizations such as the Naval Research Laboratory, Westinghouse Electric Corp., and General Electric Co.

Pursuant to the Atomic Energy Act of 1954, as amended, and section 553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions for consideration in connection with the proposed amendments should send them to the Secretary of the Commission, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments received may be examined at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

1. In § 50.55a, the introductory language in paragraph (b) is amended, paragraphs (h) and (i) are redesignated as paragraphs (i) and (j), respectively, the reference to paragraph (h) in paragraph (b)(1) is amended to refer to paragraph (i), and a new paragraph (h) is added to read as follows:

§ 50.55a Codes and standards.

Each construction permit for a utilization facility shall be subject to the following conditions, in addition to those specified in § 50.55:

* * * * *

(b) As a minimum, the systems and components of boiling and pressurized water cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (h) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (i) of this section, except as authorized by the Commission upon demonstration by the applicant for or holder of a construction permit that:

* * * * *

(h) Fracture toughness requirements: For construction permits issued on or after January 1, 1971, ferritic materials of pressure-retaining components of the reactor coolant pressure boundary shall meet the requirements set forth in Appendices G and H to this part.

* * * * *

2. New Appendices G and H are added to read as follows:

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 in order to provide adequate margins of safety under normal reactor operating conditions, system hydrostatic tests, and during transient conditions to which the system may be subjected over its service lifetime.

These requirements apply to carbon and low alloy ferritic steels (including welds and weld heat-affected zones in such materials) whose specified minimum yield strength, as defined in section II.B, does not exceed 50,000 p.s.i. Adequacy of fracture toughness of ferritic materials with higher specified minimum yield strength shall be demonstrated to the commission on an individual case basis.

II. DEFINITIONS

A. "System hydrostatic tests" mean those pressurization cycles to which the reactor coolant pressure boundary, or portions thereof, will be subjected during all hydrostatic tests of the system. Such tests include those required to comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code—Section XI—"Rules for Inservice Inspection of Nuclear Reactor Coolant Systems" as well as tests conducted prior to initial and subsequent plant startups.

B. "Specified minimum yield strength" is the minimum yield strength in the unirradiated condition of a material specified in the rules of the construction code under which the component is built, pursuant to § 50.55a.

C. "Lowest pressurization temperature" of a component is the lowest temperature at which coolant pressure within the component exceeds 25 percent of the reactor coolant system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50° F./hr., during normal reactor operation, system hydrostatic tests or transient conditions.

D. "Adjusted fracture energy" is the fracture energy of ferritic material, at a given temperature, obtained from the Charpy V-notch curve adjusted in accordance with paragraph III.B.1.

E. "Beltline region of reactor vessel" comprises the shell material, including welds and weld heat-affected zones, which directly surrounds the effective height of the fuel element assemblies, and any additional height of shell material for which the predicted shift of the Charpy V-notch (C_u) fracture energy curve exceeds 100° F.

F. "Material surveillance program" means the provisions for the placement of reactor vessel 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 vessel as a result of neutron irradiation.

G. "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 fracture toughness requirements of section IV.A, both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch impact test specified by the American Society for Testing and Materials (ASTM A-370). In addition, unirradiated ferritic materials shall be tested by means of the dropweight test (ASTM E-208). Charpy V-notch impact tests shall be conducted in accordance with the following requirements, and the adjusted fracture energy levels determined as specified in section III.B:

1. Charpy V-notch (C_v) impact tests shall be conducted to define the C_v test curve (including the upper-shield energy level) using Type A specimens oriented with respect to the "weak" direction (WR orientation in plates) of plates, forgings, castings, pipe, and tubes intended for pressure-retaining components.

2. In lieu of the specimens specified in section III.A.1, C_v specimens oriented with respect to the "strong" direction (RW orientation in plates) may be used provided test correlation data obtained from ferritic materials of the same specification are available to convert the C_v test curve (RW orientation) to the C_v test curve (WR orientation).

3. In lieu of the requirement of section III.A.2, C_v specimens oriented with respect to the "strong" direction may be used to demonstrate adequate fracture toughness provided that materials exhibit, at the lowest pressurization temperature, adjusted fracture energy levels no lower than two times the energy levels of section IV.A.

4. Materials used to prepare test specimens shall be representative of the actual properties 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 requirements of section III.A.5.

5. Materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from excess material and welds in the vessel shall course(s) following completion of the production longitudinal weld joint, and subjected to the heat treatment equivalent to that received by the vessel throughout its fabrication process. Where seamless shell forgings are used, the test specimens shall be taken from a separate weldment using excess material from the shell forging(s) and welded under the same production welding conditions applied in joining the shell forgings.

6. Charpy V-notch impact test machines used to determine fracture toughness properties for comparison with the criteria of sections IV.A and IV.B shall have been calibrated at least once in each 6-month interval using methods outlined in ASTM E23-60, and employing standard specimens obtained from U.S. Army Materials Research Center.

7. Temperature instrumentation used to control test temperature of specimens, for both Charpy V-notch impact tests and dropweight tests, shall have been calibrated at least once in each 3-month interval.

8. Persons 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 or the licensee.

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

(b) The tests have been conducted using machines and instrumentation with available records of periodic calibration, and

(c) Records of the qualifications of the individuals performing the tests are available upon request.

B. Adjusted fracture energy:

The Charpy V-notch (C_v) test curve as derived from the tests in section III.A shall be adjusted to establish the adjusted fracture energy of each material tested and to determine compliance with the acceptance requirements specified in section IV.A as follows:

1. The Charpy V-notch curve of paragraph III.A shall be translated to the right along the temperature coordinate by a temperature increment equal to the sum of:

(a) The difference between the Nil-Ductility Transition (NDT) temperature derived from the dropweight test (DWT), and the temperature corresponding to a Charpy V-notch energy value of 15 ft.-lbs. as obtained from tests on unirradiated specimens (to be applied only when the NDT temperature is higher than the temperature corresponding to the 15 ft.-lbs. Charpy V-notch energy), and

(b) A "size-effect" increment of 7° F. per inch, or fraction thereof, of material thickness.

2. The adjusted fracture energy, as read from the adjusted C_v curve of section III.B.1 at the lowest pressurization temperature, shall be used to determine compliance with the fracture toughness requirement of section IV.A.

IV. FRACTURE TOUGHNESS REQUIREMENTS

A. Ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (except as qualified under section IV.E) shall exhibit throughout their service lifetime, at the lowest pressurization temperature, adjusted fracture energy levels no lower than the following:

Section thickness t (inches):	*Minimum Charpy V-notch adjusted fracture energy (ft.-lbs.)*
$t \geq 5$	[1] 50
$2 \leq t < 5$	45
$t < 2$	40

[1] For reactor vessel beltline region this minimum fracture energy level may be inadequate for plates and forgings thicker than 12 inches. The proposed minimum fracture toughness for such vessels shall be subject to review and approval by the Commission on an individual case basis.

B. The initial upper shelf fracture energy levels, as determined by Charpy V-notch tests, shall be at least 15 ft.-lbs. higher than the values specified under section IV.A, except for reactor vessel beltline material which shall meet the additional requirements of section IV.C.

C. For the reactor vessel beltline region the upper shelf fracture energy levels for unirradiated material, as determined by Charpy V-notch tests, shall meet the following requirements, except where it can be conservatively demonstrated to the Commission by appropriate data and analyses that lower values of upper shelf fracture energy are adequate.

1. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels, and making proper allowances for all uncertainties in the measurements, that the adjusted fracture energy level of the reactor vessel beltline region will meet the requirements of section IV.A at a temperature of 100° F., over the entire service lifetime of the reactor vessel, the upper shelf fracture energy levels for unirradiated material shall meet the requirements of section IV.B.

2. For reactor vessels which do not meet the conditions of section IV.C.1 but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted fracture energy levels of the reactor vessel beltline region will meet the requirements of section IV.A at a temperature of 200° F., over the service lifetime of the reactor vessel, the upper shelf fracture energy levels for unirradiated material shall be at least 20 ft.-lbs. higher than the values specified in section IV.A.

3. For reactor vessels which do not meet the conditions of section IV.C.2, the upper shelf fracture energy levels for unirradiated material shall be at least 25 ft.-lbs. higher than the values specified in section IV.A.

D. Reactor vessels which do not meet the conditions of section IV.C.2 shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

E. Ferritic material one-half inch and less in thickness, when made to fine-grain practice, may be used in pressure retaining components of the reactor coolant pressure boundary without compliance with the requirements of section IV.A provided their lowest pressurization temperature is not less than 100° F.

V. INSERVICE REQUIREMENT—REACTOR VESSEL BELTLINE MATERIAL

A. Reactor vessels shall have their beltline region materials and weld properties monitored by a material surveillance program conforming to the "Reactor Vessel Material Surveillance Program Requirements", set forth in Appendix H.

B. Reactor vessels shall be acceptable for continued operation for that service period within which the predicted adjusted fracture energy, at the lowest pressurization temperature (as predicted from the test results of the material surveillance program of section V.A.), satisfies the requirements of section IV.A.

C. In the event that the requirements of section IV.A cannot be satisfied, reactor vessels are acceptable for continued operation provided the following requirements are satisfied for the specified conditions:

1. If the predicted adjusted fracture energy level is not less than 35 ft.-lbs., the beltline region of the vessel shall be subjected to essentially 100 percent volumetric examination in accord with the rules of ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems," section XI, and a fracture mechanics analysis shall be performed which conservatively demonstrates, making proper allowances for all uncertainties in the measurements, that adequate safety margins exist for continued operation. Such analysis shall be based on:

(a) Flaw sizes detected by the inservice inspection,

(b) Valid fracture toughness data (as defined by: "Tentative Method of Test for Plane-Strain Fracture Toughness of Metallic Materials," ASTM Designation: E 399-70T) for the base metal, welds, and weld heat-affected zones, irradiated to a level equivalent to that of the reactor vessel beltline region, and

(c) Stress analyses of the beltline region.

2. If the predicted adjusted fracture energy level is lower than 35 ft.-lbs., the reactor vessel beltline region shall be subject to a thermal annealing treatment to effect recovery of material toughness properties. The degree of such recovery shall be monitored by testing specimens from the surveillance program capsules before and after annealing treatment, and shall be adequate to satisfy the requirements of section IV.A at the end of the proposed service period.

3. If the requirements of section V.C. 1 or 2 cannot be satisfied, the licensee shall demonstrate, by other appropriate means, that adequate safety margins exist for continued operation.

The proposed programs for satisfying the requirements of section V.C. 1, 2, or 3, 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 energy levels will no longer satisfy the requirements of section IV.A.

APPENDIX H—REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I. INTRODUCTION

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water cooled power reactors as a consequence of neutron irradiation. Under this program, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel which will permit determining the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

II. SURVEILLANCE PROGRAM CRITERIA

Reactor vessels constructed of ferritic materials shall have their beltline region monitored by a surveillance program complying with the practice recommended by the American Society for Testing and Materials (ASTM) in "Surveillance Tests on Structural Materials in Nuclear Reactors", ASTM Designation: E 185-70, except as modified by the following requirements:

A. Surveillance specimens shall be taken directly from the excess shell course material, welds, and heat-affected zones of the beltline region of the reactor vessel, which are used to conduct the fracture toughness tests in meeting the requirements of section IV of Appendix G. The specimen type shall comply with the requirements of section III.A of Appendix G.

B. Irradiation capsules containing the surveillance specimens shall be located as close as practicable to the inside vessel wall, but shall not be attached to the wall. In any case, the capsule locations shall be such that the calculated neutron flux received by the innermost (with respect to the reactor core) irradiation specimens will not exceed three times the calculated maximum neutron flux at the inside wall of the vessel. The design and location of the capsules shall permit insertion of replacement capsules.

C. The required number of capsules and their withdrawal schedule are as follows:

1. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels, and making proper allowances for all uncertainties in the measurements, that the adjusted fracture energy level of the reactor vessel beltline region will meet the requirements of section IV.A of Appendix G at a temperature of 100° F. over the service lifetime of the reactor vessel, at least three capsules shall be provided for subsequent withdrawal as follows:

	Withdrawal schedule
First capsule	One-fourth service life.
Second capsule	Three-fourth service life.
Third capsule	Standby.

In the event the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the Charpy V-notch (C_v) fracture energy curve greater than originally predicted by test data, the remaining withdrawal schedule shall be modified as follows:

	Revised withdrawal schedule
Second capsule	One-half service life.
Third capsule	Standby.

2. For reactor vessels which do not meet the conditions of section II.C.1 but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted fracture energy levels of the reactor vessel beltline region will meet the requirements of section IV.A of Appendix G, at a temperature of 200° F. over the service lifetime of the reactor vessel, at least four capsules shall be provided for the subsequent withdrawal as follows:

	Withdrawal schedule
First capsule	At the time when predicted shift of C_v adjusted fracture energy curve is approximately 50° F. or at one-fourth service life, whichever is earlier.
Second capsule	At approximately one-half of the time interval between first and third capsule withdrawal.
Third capsule	Three-fourths service life.
Fourth capsule	Standby.

3. For reactor vessels which do not meet the conditions of section II.C.2, at least five capsules shall be provided for subsequent withdrawal as follows:

	Withdrawal schedule
First capsule	At the time when predicted shift of C_v adjusted fracture energy curve is approximately 50° F. or at one-fourth service life, whichever is earlier.
Second and third capsules.	At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.
Fourth capsule	Three-fourths of service life.
Fifth capsule	Standby.

4. Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

5. Sufficient archive material shall be retained to prepare additional surveillance specimens (as recommended by ASTM Designation: E 185-70 "Surveillance Tests on Structural Materials in Nuclear Reactors") except for reactor vessels which meet the conditions of section II.C. 1 or 2. The archive material shall be obtained from the excess shell course material, welds, and heat-affected zone as identified in section II.A.

III. INTEGRATED SURVEILLANCE PROGRAM

A. For multiple reactors located at a single site, each of which meets the conditions of section II.C.1, the minimum surveillance program requirements of section II.C.1 shall be met for each reactor.

B. For multiple reactors located at a single site, each of which meets the conditions of section II.C.2, an integrated surveillance program may be employed, provided that:

1. All reactor vessels meet the following additional conditions:

(a) The reactor vessels are of the same design, ordered to the same design specification, and constructed by the same fabricator using the materials produced to the same specifications, and employing the same fabrication procedures.

(b) All reactors will be operated under comparable conditions and service.

(c) Each vessel contains material specimens obtained from its respective beltline region as required by the provisions of sections II.A.

(d) The most conservative value of adjusted fracture energy levels determined from tests of specimens withdrawn from any of the reactors will be applied to all reactor vessels in establishing operational limitations.

2. The required number of capsules and their withdrawal schedule are as follows:

(a) At least four capsules for each vessel shall be provided for subsequent withdrawal.

(b) The withdrawal schedule for the vessel initially placed in service shall correspond to the schedule specified in section II.C.2.

(c) The withdrawal schedule for the other vessels shall correspond approximately to the schedule for the withdrawal of the last two capsules from the vessel initially placed in service, and the remaining two capsules shall be retained as standbys.

C. For multiple reactors located at a single site, which do not meet the conditions of section II.C.2, an integrated surveillance program may not be employed.

IV. FRACTURE TOUGHNESS TESTS

A. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of section III of Appendix G, "Fracture Toughness Requirements."

B. The test results shall be adjusted in accordance with the procedure specified under section III of Appendix G to verify that the fracture toughness requirements of section IV.A of Appendix G are satisfied.

V. REPORT OF TEST RESULTS

A. Each specimen withdrawal and the fracture toughness test shall be the subject of a summary technical report to be provided to the Commission. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the translation of the measured results to those expected in the reactor vessel beltline region.

B. The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values.

C. The lowest pressurization temperature established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes in operational procedures which are adopted to assure meeting such temperature limitations.

(Sec. 161, 68 Stat. 948, 42 U.S.C. 2201)

Dated at Washington, D.C., this 4th day of June 1971.

For the Atomic Energy Commission.

W. B. McCOOL,
Secretary of the Commission.

[FR Doc.71-9453 Filed 7-2-71;8:47 am]

June 1, 1973 SECY-R 700

CONSENT CALENDAR ITEM

For: The Commissioners

Thru: Director of Regulation

Subject: AMENDMENT TO 10 CFR PART 50: APPENDIX G, "FRACTURE TOUGHNESS REQUIREMENTS," AND APPENDIX H, "REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS"

Purpose: To recommend publication in effective form of amendments to 10 CFR Part 50 which would specify requirements for the fracture toughness of the materials in the reactor coolant pressure boundary and also the requirements for a surveillance program to monitor changes in fracture toughness of the materials in the reactor vessel beltline resulting from exposure to neutron irradiation.

Discussion: Present requirements for fracture toughness and for surveillance of irradiation damage of the materials for the reactor coolant pressure boundary are covered by General Design Criteria 31 and 32 of Appendix A to 10 CFR 50. Construction of the affected components is governed by Section III of the ASME Boiler and Pressure Vessel Code as required by section 50.55a of 10 CFR 50. These amendments give specific requirements for design and operation, many of which are in the form of a reference to pertinent sections of the ASME Code with certain supplemental requirements. Some of these add fracture toughness requirements on the material, some require higher temperature before the pressure is allowed to approach operating pressure and before the core is allowed to go critical, and some are inservice requirements that are needed to cope with damage from neutron irradiation.

These amendments have the same scope as those published for comment on July 3, 1971. They provide comparable margins of safety over the critical temperature range where fracture is a possibility. However, the language and technical approach have been updated. Publication of the proposed rule occurred at a time when the fracture toughness requirements of the ASME Code were being modernized. Many of the comments [illegible] that the proposed rule be consistent with the forthcoming revisions to the ASME Code, and this has been done. Attachment A gives a comparative text of the proposed rules and the rules in their effective form. Attachment B summarizes the comments received.

Recommendations: (a) Approve the enclosed Notice of Rule Making which would amend 10 CFR Part 50 to incorporate the rules for fracture toughness requirements and for reactor vessel material surveillance program requirements, and

(b) Note:

1) The amendments will be published in the Federal Register to be effective thirty (30) days after publication;

2) The JCAE will be informed; and

3) A public announcement will be issued.

Coordination: The Directorates of Licensing and Regulatory Operations and the Office of the General Counsel concur, and the Advisory Committee on Reactor Safeguards has approved publication in effective form.

Scheduling: (a) Approvals or comments by June 11, 1973.

(b) For affirmation at an early Policy Session.

Lester Rogers

Lester Rogers
Director of Regulatory Standards

Contact:
P. N. Randall, Ext. 7546

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Attachment A

NOTICE OF RULE MAKING

TITLE 10 - ATOMIC ENERGY

CHAPTER 1 - ATOMIC ENERGY COMMISSION

PART 50 - LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Fracture Toughness and Surveillance Program Requirements

On July 3, 1971, the Atomic Energy Commission published in the FEDERAL REGISTER (36 FR 12697) proposed amendments to its regulations in 10 CFR Part 50 which would add new appendices entitled, "Appendix G, Fracture Toughness Requirements," and "Appendix H, Reactor Vessel Material Surveillance Program Requirements."

Interested persons were invited to submit written comments within 60 days. Upon consideration of the comments received and other factors involved, the Commission has adopted the proposed amendments with certain modifications in the form set forth below.

Significant differences in Appendix G from the amendments published for comment are:

(1) Terminology was changed to be consistent with that of the ASME Code.[1/]

(2) The method of combining the results of the Charpy and dropweight tests to get a combined measure of toughness was changed.

1/ American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, "Rules for the Construction of Nuclear Power Plant Components," 1971 Edition, and addenda through the Winter, 1972 Addenda.

The proposed rule would have required characterization of the fracture toughness of the ferritic materials in the reactor coolant pressure boundary in terms of the temperature dependence of two quantities: (a) energy absorbed in Charpy V-notch impact tests (ASTM[2/] Standard A-370) and (b) the nil-ductility transition (NDT) temperature obtained from dropweight tests (ASTM Standard E-208). Charpy tests were to be run at appropriate temperatures to characterize the transition from fully ductile, "upper shelf," behavior to low-energy, "brittle," behavior. To obtain a toughness characterization that depended on both types of tests, the "Charpy curve" was to be adjusted upward on the temperature scale to make the 15 ft. lb. level correspond to the NDT temperature from the dropweight tests.

These amendments continue the requirement contained in the proposed rule that fracture toughness be measured by the Charpy test and the dropweight test. However, to reflect comments urging consistency with the ASME Code, fracture toughness of the material is characterized by its reference temperature, RT_{NDT}. This temperature is the higher value of the NDT temperature from the dropweight test or the temperature that is 60°F below the temperature at which Charpy test data meet a specified toughness level (50 ft. lbs. and 35 mils lateral expansion).

2/ American Society for Testing and Materials.

(3) The concept of a lowest pressurization temperature given in the proposed rule was changed to a concept based on fracture mechanics that allows a continuous buildup of pressure as a function of temperature and wall thickness.

The proposed rule would have required a "thickness correction" whereby the Charpy curve was to be shifted up the temperature scale 7°F per inch of material thickness. The thickness correction would have been added to the shift required for consistency between the two types of toughness tests to obtain a curve of "adjusted fracture energy" versus temperature. Fracture control would have been achieved by requiring the "lowest pressurization temperature" at which system pressure could exceed 25 percent of normal operating pressure, or at which the rate of temperature change could exceed 50°F/hr., to be the temperature at which the adjusted fracture energy exceeded a certain level, which was higher for thick material than for thin.

Many of the comments questioned the validity of the dependence placed on the Charpy test by the proposed rule. The thickness correction was considered excessive for thick sections and inadequate for thin sections. Other comments asked that the rules treat stresses more quantitatively to take account of the operators' ability to control pressure and rate of temperature change and the designers' ability to calculate pressure and thermal stresses. Specifically, they urged the adoption of the approach that now appears in the 1972 Summer Addenda to the ASME Code. The proposed rules were also revised to reflect these comments.

As required by these amendments, fracture control is achieved by requiring that stress in the pressure boundary be limited as a function of the metal temperature relative to the reference temperature, RT_{NDT}, and as a function of material thickness according to the "K_{IR} curve" given in the ASME Code. Taken from fracture mechanics, the term "stress intensity factor" (K) defines a quantity that is proportional to the product of gross stress and the square root of crack depth, and includes factors to account for crack shape and for the manner of loading. Critical values of K, determined from tests in which precracked specimens are loaded to failure, are a convenient measure of fracture toughness, because differences in crack size and shape and differences in manner of loading between specimen and component can be treated quantitatively. The K_{IR} curve in the ASME Code gives allowable values of fracture toughness as a function of temperature relative to RT_{NDT}. The curve is based on data obtained from tests of large specimens in the HSST[3/] program. Rather than require the estimation of maximum expected flaw size, these amendments require that in areas of the reactor vessel

3/ Heavy Section Steel Technology Program, conducted at Oak Ridge National Laboratory.

remote from discontinuities, the assumed flaw size be proportional to wall thickness. Thus, from the value of K_{IR} at a given temperature, allowable stress values are obtained that are inversely proportional to the square root of wall thickness.

(4) Fracture control procedures described in paragraph (3), above, are supplemented in these amendments by a requirement that whenever the core is critical, the metal temperature of the reactor vessel shall exceed specified values dependent on the concurrent stress level.

(5) The Charpy V-notch upper-shelf energy requirements for beltline region materials was set at 75 ft. lbs. for all cases, without distinction as to the predicted amount of irradiation damage.

(6) Fracture toughness requirements for the various components of the pressure boundary were separated to reflect comments suggesting that the rules fit the anticipated severity of service to which the component might be subjected.

(7) The definition of "beltline region of the reactor vessel" was broadened to include more shell material above and below the core.

Significant differences in Appendix H from the amendments published for comment are:

(1) Terminology was changed to be consistent with that of Appendix G and the ASME Code. In particular, the adjustment for irradiation effects is described in these amendments as an adjustment of the reference temperature, RT_{NDT}, and the amount of temperature shift is determined by a slightly different treatment of the Charpy data than that given in the proposed amendment.

(2) Provision was made for accelerated irradiation capsules and for modification of capsule withdrawal schedules based on results of tests of specimens that received the accelerated irradiation.

(3) A general provision for an integrated surveillance program was substituted for the specific requirements given in the proposed rule. It appeared from comments that it would be impractical to meet the requirements of the proposed rule for commonality of multiple reactors.

Appendices G and H are intended to implement General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," to the extent described below. The margin of safety against brittle fracture will be controlled more quantitatively by these amendments than by the proposed rule, particularly with regard to specific guidelines for the treatment of heatup and cooldown conditions. Appendices G and H track the language of the ASME Code and have adopted certain of its requirements but also include several key supplemental requirements. For the vessel beltline, inservice requirements are based on the reference temperature as adjusted to account for irradiation damage. There is also an additional fracture toughness requirement in the form of shelf energy values from the Charpy curve for the material in its unirradiated condition.

Although the requirements of Appendices G and H become effective on ________________, the Commission recognizes that there may be an interim period when, for plants now under construction, the method of compliance with certain provisions may be determined on a case-by-case basis. For example, if the test data needed to establish certain fracture control requirements are not available because they were not required at the time material sampling was done, estimated values that are appropriately conservative may be acceptable.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Part 50, are published as a document subject to codification to be effective on ____________________. [30 days after publication in the FEDERAL REGISTER]

1. In §50.55a of 10 CFR Part 50, a new paragraph (i) is added and subdivision (a)(2)(i) and the prefatory language in paragraph (a)(2) are amended to read as follows:

§50.55a Codes and standards.

Each construction permit for a utilization facility shall be subject to the following conditions, in addition to those specified in §50.55:

(a)(1) * * *

(2) As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (i) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing, or inspection of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety; or

(ii) * * *

* * * * *

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 [~~in-order~~] to provide adequate margins of safety [~~under-normal-reactor-operating-conditions, system-hydrostatic-tests, and-during-transient-conditions-to-which-the system-may-be-subjected-over-its-service-lifetime.~~] 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.

[~~These-requirements-apply-to-carbon-and-low-alloy-ferritic-steels-(including welds-and-weld-heat-affected-zones-in-such-materials)-whose-specified-minimum yield-strength,-as-defined-in-section-II.B,-does-not-exceed-50,000-p.s.i. Adequacy-of-fracture-toughness-of-ferritic-materials-with-higher-specified minimum-yield-strength-shall-be-demonstrated-to-the-commission-on-an---individual-case-basis.~~] 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.

[A]C. "System hydrostatic tests" means [~~those-pressurization-cycles-to which-the-reactor-coolant-pressure-boundary,-or-portions-thereof, will-be-subjected-during-all-hydrostatic-tests-of-the-system.--Such~~

[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.

~~tests-include-those-required-to-comply-with-the-American-Society-of Mechanical-Engineers-(ASME)-Boiler-and-Pressure-Vessel-Code~~] 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." [~~as-well-as-tests-conducted-prior-to initial-and-subsequent-plant-startups.~~]

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

[C. ~~"Lowest-pressurization-temperature"-of-a-component-is-the-lowest-temperature-at-which-coolant-pressure-within-the-component-exceeds-25-percent of-the-reactor-coolant-system-normal-operating-pressure,-or-at-which-the rate-of-temperature-change-in-the-component-material-exceeds-50°F./hr., during-normal-reactor-operation,-system-hydrostatic-tests-or-transient conditions.~~]

[D. ~~"Adjusted-fracture-energy"-is-the-fracture-energy-of-ferritic-material, at-a-given-temperature,-obtained-from-the-Charpy-V-notch-curve-adjusted in-accordance-with-paragraph-III.B.1.~~]

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.

[E]H. "Beltline region of reactor vessel" [~~comprises~~] 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 [~~shift-of-the-Charpy-V-notch-(Cv)-fracture-energy-curve~~] adjustment of reference temperature at end of service life of the reactor vessel exceeds [~~100°~~] 50°F.

[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 [~~vessel~~] as a result of exposure to neutron irradiation and the thermal environment.

[6]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 fracture toughness requirements of section IV.A, both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch impact test specified by the American Society for Testing and Materials (ASTM A 370). In addition, unirradiated ferritic materials shall be tested by means of the drop weight test (ASTM E 208). Charpy V-notch impact tests shall be conducted in accordance with the following requirements, and the adjusted fracture energy levels determined as specified in section III.B:~~

1. ~~Charpy V-notch (Cv) impact tests shall be conducted to define the Cv test curve (including the upper shelf energy level) using Type A specimens oriented with respect to the "weak" direction (WR orientation in plates) of plates, forgings, castings, pipe, and tubes intended for pressure retaining components.~~

2. ~~In-lieu-of-the-specimens-specified-in-section-III.A.1,-Cv-specimens-oriented-with-respect-to-the-"strong"-direction-(RW-orientation-in-plates)-may-be-used-provided-test-correlation-data-obtained-from-ferritic-materials-of-the-same-specification-are-available-to-convert-the-Cv-test-curve-(RW-orientation)-to-the-Cv-test-curve-(WR-orientation).~~

3. ~~In-lieu-of-the-requirement-of-section-III.A.2,-Cv-specimens-oriented-with-respect-to-the-"strong"-direction-may-be-used-to-demonstrate-adequate-fracture-toughness-provided-that-materials-exhibit,-at-the-lowest-pressurization-temperature,-adjusted-fracture-energy-levels-no-lower-than-two-times-the-energy-levels-of-section-IV.~~]

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.

[A4.]2. Materials used to prepare test specimens shall be representative of the actual [~~properties~~] 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.A.5.~~] III.C. of this appendix.

[A6. ~~Charpy-V-notch-impact-test-machines-used-to-determine-fracture-toughness-properties-for-comparison-with-the-criteria-of-sections-IV.A-and-IV.B-shall-have-been-calibrated-at-least-once-in-each-6-month-interval-using-methods-outlined-in-ASTM-E23-60,-and-employing-standard-specimens-obtained-from-U.S.-Army-Materials-Research-Center~~.

A7. ~~Temperature-instrumentation-used-to-control-test-temperature-of--specimens,-for-both-Charpy-V-notch-impact-tests-and-drop-weight-tests,-shall-have-been-calibrated-at-least-once-in-each-3-month-interval~~.]

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.

[A8.]4. [~~Persons~~] 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. [~~of the licensee.~~]

[A9.]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,

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

[~~(b)~~]c. The tests have been conducted using machines and instrumentation with available records of periodic calibration, and

[~~(c)~~]d. Records of the qualifications of the individuals performing the tests are available upon request.

B. ~~Adjusted fracture energy:~~

~~The Charpy V-notch (Cv) test curve as derived from the tests in section III.A shall be adjusted to establish the adjusted fracture energy of each material tested and to determine compliance with the acceptance requirements specified in section IV.A as follows:~~

1. ~~The Charpy V-notch curve of paragraph III.A. shall be translated to the right along the temperature coordinate by a temperature increment equal to the sum of:~~

   (a) ~~The difference between the Nil Ductility Transition (NDT) temperature derived from the dropweight test (DWT), and the temperature corresponding to a Charpy V-notch energy value of 15 ft. lbs. as obtained from tests on unirradiated specimens (to be applied only when the NDT temperature is higher than the temperature corresponding to the 15 ft. lbs. Charpy V-notch energy), and~~

   (b) ~~A "size effect" increment of 7°F. per inch, or fraction thereof, of material thickness.~~

2. ~~The adjusted fracture energy, as read from the adjusted Cv curve of section III.B.1 at the lowest pressurization temperature shall be used to determine compliance with the fracture toughness requirement of section IV.A.~~]

C. In addition to the test requirements of section III.A. of this appendix, tests on materials of the reactor vessel beltline shall be conducted in accordance with the following minimum requirements:

1. Charpy V-notch (Cv) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the Cv 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.

[~~A5~~.]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 [~~the~~] a heat treatment that produces metallurgical effects equivalent to [~~that~~] those [~~received by~~] 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 [~~shall~~] 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 [~~welded under~~] the same production welding conditions [~~applied~~] as those used in joining the corresponding shell [~~forgings~~] materials.

IV. FRACTURE TOUGHNESS REQUIREMENTS

[A. ~~Ferritic materials of pressure retaining components of the reactor coolant pressure boundary (except as qualified under section IV.E) shall exhibit throughout their service lifetime, at the lowest pressurization temperature, adjusted fracture energy levels no lower than the following~~:

~~Section thickness t~~ (~~inches~~):	~~Minimum Charpy V-notch adjusted fracture energy~~ (~~ft.-lbs~~)
~~t≥5~~	[1] ~~50~~
~~2≤t<5~~	~~45~~
~~t<2~~	~~40~~

[1] ~~For reactor vessel beltline region this minimum fracture energy level may be inadequate for plates and forgings thicker than 12 inches. The proposed minimum fracture toughness for such vessels shall be subject to review and approval by the Commission on an individual case basis.~~]

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 pre-operational 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. The initial upper shelf fracture energy levels, as determined by Charpy V-notch tests, shall be at least 15 ft. lbs higher than the values specified under section IV.A. except for reactor vessel beltline material which shall meet the additional requirements of section IV.C.~~

~~C. For the reactor vessel beltline region the upper shelf fracture energy levels for unirradiated material, as determined by Charpy V notch tests, shall meet the following requirements, except where it can be conservatively demonstrated to the Commission by appropriate data and analyses that lower values of upper shelf fracture energy are adequate.~~

~~1. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels, and making proper allowances for all uncertainties in the measurements, that the adjusted fracture energy level of the reactor vessel beltline region will meet the requirements of section IV.A at a temperature of 100°F, over the entire service lifetime of the reactor vessel, the upper shelf fracture energy levels for unirradiated material shall meet the requirements of section IV.B.~~

~~2. For reactor vessels which do not meet the conditions of section IV.G.1 but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted fracture energy levels of the reactor vessel beltline region will meet the requirements of section IV.A at a temperature of 200°F., over the service lifetime of the reactor vessel, the upper shelf fracture energy levels for unirradiated material shall be at least 20 ft. lbs. higher than the values specified in section IV.A.~~

~~3. For reactor vessels which do not meet the conditions of section IV.G.2. the upper shelf fracture energy levels for unirradiated material shall be at least 25 ft. lbs. higher than the values specified in section IV.A.~~]

B. Reactor vessel 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.

[B]C. Reactor vessels ~~[which-do-not-meet-the-conditions-of-sections-IV.C.2]~~ 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.

[~~E.--Ferritic-material-one-half-inch-and-less-in-thickness,-when-made-to fine-grain-practice,-may-be-used-in-pressure-retaining-components-of the-reactor-coolant-pressure-boundary-without-compliance-with-the requirements-of-section-IV.A-provided-their-lowest-pressurization temperature-is-not-less-than-100°F.~~]

V. INSERVICE REQUIREMENTS--REACTOR VESSEL BELTLINE MATERIAL

A. [~~Reactor-vessels-shall-have-their~~] The properties of reactor vessel beltline region materials, [~~and-weld-properties~~] 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 [~~shall-be-acceptable-for-continued-operation~~] may continue to be operated only for that service period within which the [~~predicted-[adjusted-fracture-energy,-at-the-lowest-pressurization~~

~~temperature-(as-predicted-from-the-test-results-of-the-material surveillance-program-of-section-V.A.),-satisfies-the-requirements-of section-IV.A.]~~ 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 [~~IV.A~~] V.B. cannot be satisfied, reactor vessels [~~are-acceptable-for-continued-operation~~] may continue to be operated provided all of the following requirements are satisfied [~~for-the-specified-conditions~~]:

~~1.--If-the-predicted-adjusted-fracture-energy-level-is-not-less-than 35-ft.-lbs.,-the-beltline-region-of-the-vessel-shall-be-subjected to-essentially-100-percent-volumetric-examination-in-accord-with-the rules-of-ASME-Boiler-and-Pressure-Vessel-Code,-"Rules-for-Inservice Inspection-of-Nuclear-Reactor-Coolant-Systems,"-section-XI,-and-a fracture-mechanics-analysis-shall-be-performed-which-conservatively demonstrates,-making-proper-allowances-for-all-uncertainties-in-the measurements,-that-adequate-safety-margins-exist-for-continued operation.--Such-analysis-shall-be-based-on:~~

~~(a)--Flaw-sizes-detected-by-the-inservice-inspection,~~

~~(b)--Valid-fracture-toughness-data-(as-defined-by:--"Tentative Method-of-Test-for-Plane-Strain-Fracture-Toughness-of Metallic-Materials,"--ASTM-Designation:--E-399-70T)-for the-base-metal,-welds,-and-weld-heat-affected-zones, irradiated-to-a-level-equivalent-to-that-of-the-reactor vessel-beltline-region,-and~~

~~(c)--Stress-analyses-of-the-beltline-region.~~

1. An essentially 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 safety margins for continued operation.

~~[2. If the predicted adjusted fracture energy level is lower than 35 ft-lbs, the reactor vessel beltline region shall be subject to a thermal annealing treatment to effect recovery of material toughness properties. The degree of such recovery shall be monitored by testing specimens from the surveillance program capsules before and after annealing treatment, and shall be adequate to satisfy the requirements of section IV.A at the end of the proposed service period.~~]

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.

[~~3. If the requirements of section V.C.1 or 2 cannot be satisfied, the licensee shall demonstrate, by other appropriate means, that adequate safety margins exist for continued operation.~~]

E. The proposed programs for satisfying the requirements of sections [~~V.C.1, 2, or 3,~~] 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 [~~energy~~] toughness levels will no longer satisfy the requirements of section [~~IV.A.~~] V.B.

APPENDIX H--REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I. INTRODUCTION

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water cooled power reactors [~~as a consequence of~~] resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. [~~which~~] These data will permit [~~determining~~] the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

II. SURVEILLANCE PROGRAM CRITERIA

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods, applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E> 1MeV) at the end of the design life of the vessel will not exceed [$\cancel{5 \times 10^{15}}$] 10^{17} n/cm^2.

B. Reactor vessels constructed of ferritic materials which do not meet the conditions of section II.A. shall have their beltline regions monitored by a surveillance program complying with the [~~practice~~

~~recommended-by-the~~] American Society for Testing and Materials (ASTM) Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, [~~in-"Surveillance-Tests-on-Structural-Materials in-Nuclear-Reactors,"~~] ASTM Designation: [~~E-185-70~~] E-185-73,[1/] except as modified by [~~the-following-requirements:~~] this appendix.

C. The surveillance program shall meet the following requirements:

{~~A.~~} 1. Surveillance specimens shall be taken {~~directly-from-the-excess shell-course-material,-welds,-and-heat-affected-zones-of-the beltline-region-of-the-reactor-vessel.~~} from locations alongside [~~which-are-used-to-conduct~~] the fracture toughness test specimens required by [~~in-meeting-the-requirements-of~~] section III of Appendix G. The specimen {~~type~~} types shall comply with the requirements of section III.A. of Appendix G (except that drop-weight specimens are not required).

{~~B.~~} 2. Surveillance ~~irradiation~~ capsules containing the surveillance specimens shall {~~be-located-as-close-as-practicable-to-the inside-vessel-wall,-but-shall-not-be-attached-to-the-wall.--In any-case,-the-capsule-locations-shall-be-such-that-the-calculated neutron-flux-received-by-the-innermost-(with-respect-to-the-reactor core)-irradiation-specimens-will-not-exceed-three-times-the~~

[1/] Effective March 1, 1973. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103, either as a separate or (when available) as part of the 1973 Annual Book of ASTM Standards, Part 30 and also in Part 31. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. N.W., Washington, D. C.

~~calculated-maximum-neutron-flux-at-the-inside-wall-of-the-vessel. The-design-and-location-of-the-capsules-shall-permit-insertion-of replacement-capsules.~~} be located near but not attached to the inside vessel wall in the beltline region, so that the neutron flux received by the specimens is at least as high but not more than three times as high as that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules, for which the calculated neutron flux will exceed three times the calculated maximum neutron flux at the inside wall of the vessel, may be used in addition to the required number of surveillance capsules specified in section II.C.3.

~~{6}~~ 3. The required number of surveillance capsules and their withdrawal schedules are as follows:

~~{1}~~ a. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steel,[~~s-and~~] making [~~proper~~] appropriate allowances for all uncertainties in the measurements, that the adjusted [~~fracture-energy-level-of-the-reactor-vessel beltline-region-will-meet-the-requirements-of-section-IV.A. of-Appendix-G-at-a-temperature-of~~] reference temperature established in accordance with section IV.B. will not exceed

100°F [~~over~~] at the end of the service lifetime of the reactor vessel, at least three surveillance capsules shall be provided for subsequent withdrawal as follows:

Withdrawal Schedule

First capsule - One-fourth service life

Second capsule - Three-fourths service life

Third capsule - Standby

In the event that the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the ~~[Charpy-V-notch (C_V) fracture energy curve greater than originally predicted by test data,]~~ reference temperature greater than orginally predicted for similar material as recorded ~~and documented~~ in the applicable technical specifications, the remaining withdrawal schedule shall be modified as follows:

Revised
Withdrawal Schedule

Second capsule - One-half service life

Third capsule - Standby

~~[2.]~~ b. For reactor vessels which do not meet the conditions of section ~~[II.C.1]~~ II.C.3.a. but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted ~~{fracture-energy levels-of-the-reactor-vessel-beltline-region-will-meet-the requirements-of-section-IV.A-of-Appendix-G,-at-a-temperature of}~~ reference temperature will not exceed 200°F [-~~over~~] at the end of the service lifetime of the reactor vessel, at least four surveillance capsules shall be provided for the subsequent withdrawal as follows:

Withdrawal Schedule

First capsule -	At the time when the predicted shift of ~~{C_V--adjusted-fracture-energy-curve}~~ the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.
Second capsule-	At approximately one-half of the time interval between first and third capsule withdrawal.
Third capsule -	Three-fourths service life.
Fourth capsule-	Standby.

~~{(3)}~~ c. For reactor vessels which do not meet the conditions of section ~~{II.C.2}~~ II.C.3.b., at least five surveillance capsules shall be provided for subsequent withdrawal as follows:

Withdrawal Schedule

First capsule -	At the time when the predicted shift of ~~{C_V-adjusted-fracture-energy-curve}~~ the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.
Second and third capsules-	At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.

Fourth capsule- Three-fourths of service life.
Fifth capsule - Standby.

d. Provision shall also be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation.

~~[4.]~~ e. Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

f. If accelerated irradiation capsules are employed in addition to the minimum required number of surveillance capsules, the withdrawal schedule may be modified, taking into account the test results obtained from testing of the specimens in the accelerated capsules. The proposed modified withdrawal schedule in such cases shall be approved by the Commission on an individual case basis.

g. Proposed withdrawal schedules that differ from those specified in paragraphs a. through f. shall be submitted, with a technical justification therefor, to the Commission for approval. The proposed schedule shall not be implemented without prior Commission approval.

4. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an

individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

[~~5.--Sufficient-archive-material-shall-be-retained-to-prepare-additional surveillance-specimens-(as-recommended-by-ASTM-Designation:-E-185-70-"Surveillance-Tests-on-Structural-Materials-in-Nuclear-Reactors") except-for-reactor-vessels-which-meet-the-conditions-of-section II.C.1-or-2.--The-archive-material-shall-be-obtained-from-the excess-shell-course-material,-welds,-and-heat-affected-zone-as identified-in-section-II.A.~~]

[~~III.--INTEGRATED-SURVEILLANCE-PROGRAM~~]

[~~A.--For-multiple-reactors-located-at-a-single-site,-each-of-which-meets the-conditions-of-section-[II.C.1]-II.C.3.(a),-the-minimum-surveillance program-requirements-of-section-[II.C.1]-II.C.3.(a)-shall-be-met for-each-reactor.~~]

[~~B.--For-multiple-reactors-located-at-a-single-site,-each-of-which-meets the-conditions-of-section-[II.C.2]-II.C.3.(b),-an-integrated surveillance-program-may-be-employed,-provided-that:~~]

[~~1.--All-reactor-vessels-meet-the-following-additional-conditions:~~

~~(a)--The-reactor-vessels-are-of-the-same-design,-ordered-to-the same-design-specification,-and-constructed-by-the-same fabricator-using-[the]-materials-produced-to-the-same-specifications,-and-employing-the-same-fabrication-procedures.~~]

[~~(b) All reactors will be operated under comparable conditions and service.~~]

[~~(c) Each vessel contains material specimens obtained from its respective beltline region as required by the provisions of section II.A.~~]

[~~(d) The most conservative value of adjusted fracture energy levels determined from tests of specimens withdrawn from any of the reactors will be applied to all reactor vessels in establishing operational limitations.~~]

[~~2. The required number of capsules and their withdrawal schedule are as follows:~~].

[~~(a) At least four capsules for each vessel shall be provided for subsequent withdrawal.~~]

[~~(b) The withdrawal schedule for the vessel initially placed in service shall correspond to the schedule specified in section II.C.2.~~]

[~~(c) The withdrawal schedule for the other vessels shall correspond approximately to the schedule for the withdrawal of the last two capsules from the vessel intially placed in service, and the remaining two capsules shall be retained as standbys.~~]

[~~C. For multiple reactors located at a single site, which do not meet the conditions of section II.C.2, an integrated surveillance program may not be employed.~~]

III. [~~IV.~~] FRACTURE TOUGHNESS TESTS

A. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of section III of Appendix G, "Fracture Toughness Requirements."

[~~B. The test results shall be adjusted in accordance with the procedure specified under section III of Appendix G to verify that the fracture toughness requirements of section IV.A of Appendix G are satisfied.~~]

<u>B. The adjusted reference temperatures for the base metal, heat-affected zone, and weld metal shall be obtained from the test results by adding to the reference temperature the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 foot-pound level or that measured at the 35 mil lateral expansion level, whichever temperature shift is greater. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements of section V.B. of Appendix G are satisfied.</u>

IV. [~~V.~~] REPORT OF TEST RESULTS

A. Each ~~{specimen}~~ capsule withdrawal and the results of the fracture toughness ~~{test}~~ tests shall be the subject of a summary technical report to be provided to the Commission. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the ~~{translation}~~ relationship of the measured results to those [~~expected in~~] predicted for the reactor vessel beltline region.

B. The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values of fluence.

C. The ~~{lowest-pressurization-temperature}~~ operating pressure and temperature limitations established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes made in operational procedures ~~which-are-adopted~~ to assure meeting such temperature limitations.

ATTACHMENT B

ANALYSIS OF COMMENTS

The following is an analysis of the salient comments received on the proposed amendment, adding Appendices G and H to 10 CFR 50. Copies of the comments are available in the Directorate of Regulatory Standards.

A. Request for Consistency with the ASME Code

Comments

1. The letter from W. G. Hoyt, Secretary of the Boiler and Pressure Vessel Committee read as follows:

   The ASME Boiler and Pressure Vessel Committee has noted with great interest the proposed revisions of 10-CFR Part 50 and the proposed Appendices G and H as published in the Federal Register of July 3, 1971. We have been studying this subject intensively for over a year with a view toward updating the present requirements of Section III of the ASME Boiler and Pressure Vessel Code in accordance with the latest developments in the technology.

   In January 1971 we requested specific recommendations on this subject from the Pressure Vessel Research Committee. Their recommendations have now been received and are being studied by our appropriate subcommittees and subgroups. It is expected that specific Code revisions will be formulated well before the end of 1971.

   Due to our meeting schedule it is not possible to submit detailed comments on your proposed requirements before September 3, as requested. We would, however, appreciate the opportunity to comment on this subject or perhaps meet with your selected representatives at a later date so that the ASME and the AEC toughness requirements can be as consistent as possible.

2. The letter from W. D. Doty, Chairman of the Pressure Vessel Research Committee read as follows:

In January 1971, the Pressure Vessel Research Committee initiated the preparation of recommendations for toughness requirements for ferritic materials in nuclear power reactors. Concentrated effort was applied to this task and a report, "Recommendations of PVRC - Toughness Requirements for Ferritic Materials," August 13, 1971, was prepared. Industry and government personnel participated in the preparation of the report.

Enclosed are two (2) copies of this report which we are submitting to the U. S. Atomic Energy Commission in response to the subject notice of July 3, 1971 in the Federal Register. The PVRC recommendations in the report are offered as replacement rules for those proposed in the Federal Register.

PVRC should be pleased to arrange for representatives of the Committee to meet with representatives of the Atomic Energy Commission for a discussion of the PVRC report, if such a discussion is desired by the Commission.

3. These comments were echoed by: Babcock and Wilcox, Combustion Engineering, General Electric, Westinghouse, and the Tennessee Valley Authority.

Staff Action

The proposed rule was revised to make full use of pertinent provisions of the ASME Code, Section III, Summer 1972 Addenda. The language of the proposed rule was also modified to be consistent with the ASME Code.

B. Objection to the Lowest-Pressurization Temperature Concept of Fracture Control and to the Basis for Establishing it

Comments

1. To limit the pressure to 25 percent of normal operating pressure and the rate of temperature change to 50F per hour until a certain temperature is reached (beyond which there are no limits based on toughness criteria) is not consistent with the designers' ability to calculate stresses in nuclear components, and leaves no basis for evaluating the extra margin achieved by such things as reduction of cooling rate. (General Electric Co., Consolidated Edison)

2. The use of a step change in allowable pressure as a function of temperature does not provide a uniform margin of safety over the temperature range, because toughness increases steadily with temperature, even below the "transition temperature." (Westinghouse)

3. The adjustment of the Charpy fracture energy curve to a higher

temperature to make the 15 ft. lb. level coincide with the NDT temperature from the dropweight test unduly penalizes the most commonly used low-alloy pressure vessel steels. (Babcock and Wilcox, Westinghouse, Consumers Power, Consilidated Edison, Effects Technology).

4. The proposed thickness correction of 7F per inch of thickness, as a second adjustment of the fracture energy curve, is excessive for thicknesses over 10 inches (as in flanges) and is too small for the lower range of thicknesses. (Babcock and Wilcox, Combustion Engineering, Westinghouse, Tennessee Valley Authority).

Staff Action

The adoption of the requirements of the ASME Code, including its non-mandatory Appendix G, represents agreement with the comments. The following discussion paraphrases the affected parts of the proposed rule and of the substitute requirements and shows how the latter accomplish the purposes intended.

The proposed rule required characterization of the fracture toughness of the material in terms of the temperature dependence of two quantities -- energy absorbed in a Charpy V-notch impact test (ASTM Standard A-370) and the nil-ductility transition (NDT) temperature in a drop-weight test (ASTM Standard E-208). In this test, a beam specimen is subjected to the impact of a falling weight to cause a running crack to propagate from a brittle weld bead on the tension face of the specimen. NDT is the highest test temperature at which the crack reaches both edges of the specimen before the latter hits the deflection stop. Charpy tests were to be run at a series of temperatures chosen to characterize the transition from fully ductile, "upper shelf," behavior to low-energy "brittle" behavior where there is very little plastic strain before fracture. To obtain a toughness characterization that depended on both types of tests, the "Charpy curve" was adjusted upward (only) on the temperature scale to make the 15 ft. lb. level correspond to the NDT temperature from the dropweight test.

The proposed rule also required consideration of the known tendency of thick sections to suffer brittle fracture more readily than thin sections by requiring a "thickness correction." For this, the Charpy curve was shifted up the temperature scale 7 degrees per inch of material thickness. The "size-effect increment" was added to the shift required for consistency between the two types of toughness tests to obtain a curve of "adjusted fracture energy" versus temperature as the characterization of the fracture toughness of the material in the intended thickness. Fracture control was achieved by a requirement of the proposed rule that the "lowest pressurization temperature," below which pressure could not exceed 25 percent of normal operating pressure nor cooling rate exceed 50 F/hr., was required to be the temperature at which the adjusted fracture energy exceeded a certain level, which was higher for thick material than for thin. Radiation damage was accounted for in terms of a measured shift in the temperature required to achieve the specified Charpy energy levels. The surveillance program required by Appendix H provided Charpy specimens for such measurements.

The rule in its effective form continues the proposed requirement that fracture toughness be characterized by both the Charpy test and the dropweight test, but in keeping with the ASME Code, the results are interpreted in terms of a reference temperature, RT_{NDT}, which is the higher of the nil-ductility transition temperature from the dropweight test and a temperature obtained from Charpy test data in a special way. (It is 60F below the temperature at which Charpy energy equals 50 ft. lbs. and back face deformation equals 35 mils lateral expansion.)

Fracture control under the new rule is achieved by a requirement that stress in the pressure boundary be limited as a function of temperature relative to the reference temperature, RT_{NDT}, and as a function of material thickness according to the "K_{IR} curve" given in the revised ASME Code. Taken from fracture mechanics, the term "stress intensity factor," (K) defines a quantity that is proportional to the product of gross stress and the square root of crack depth, and includes factors to account for crack shape and for the manner of loading. Critical values of K, obtained from tests in which precracked specimens are loaded to failure, are a convenient measure of fracture toughness, because differences in crack size and shape and in manner of loading between specimen and component can be treated quantitatively. The K_{IR} curve in the ASME Code is regarded as a lower-bound measure of the dependence of fracture toughness on temperature, relative to RT_{NDT}, for the materials of interest. The curve is based on data obtained in the HSST program.*

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*Heavy Steel Technology Program, being conducted at Oak Ridge National Laboratory.

Rather than require an independent estimate of the maximum expected flaw size in shells and heads remote from discontinuities, the ASME Code requires that the assumed flaw for a vessel of wall thickness "t" shall be a semielliptical surface crack of depth 0.25t and length 1.5t. Thus, the value of K_{IR} at a given temperature amounts to an allowable stress value that decreases with increasing wall thickness as the square root of 1/t.

With regard to fracture control in the operation of a nuclear reactor, the rule in effective form substitutes the K_{IR} curve for the proposed requirement that the system not be pressurized above 25 percent of normal operating pressure until the temperature reached the value corresponding to a specified energy level on the adjusted Charpy curve. Thus, instead of permitting a step change in allowable pressure at a certain temperature, the K_{IR} curve permits stress levels to rise at a steadily increasing rate through the temperature range, relative to RT_{NDT}.

C. <u>Objection to Amount of Testing Required</u>

<u>Comments</u>

1. Instead of treating the entire pressure boundary alike, there should be separate requirements for the various components depending on the likelihood of fracture. (Babcock and Wilcox, Consumers Power)

2. Requirements for testing fracture properties in the transverse direction are unreasonable for plates which are stressed in the rolling direction, in which fracture toughness is greater. (Combustion Engineering, Consumers Power).

Staff Action

The effective rule has been changed to accommodate comment (1) but the requirements for specimen location and orientation referred to in comment (2) were changed only to be consistent with the ASME Code. Toughness in the transverse direction is a good measure of plate quality, and quality assurance is made easier if orientation of the plate in the vessel is not an essential factor.

D. Objection to the Definition of Beltline and to the Extra Material Testing Required for Beltline Materials

1. Beltline should be defined as material receiving a predicted fluence exceeding 10^{19} n/cm^2 (E > 1MeV), rather than a certain predicted shift in temperature of the Charpy V-notch energy curve. Also, withdrawal schedules should be based on fluence rather than a "Charpy shift."

2. Charpy V-notch upper-shelf requirements should be based on the expected "Charpy shift" instead of an absolute temperature. (General Electric Co.).

Staff Action

Comment (2) has been accommodated but not comment (1). Although material characteristics are not completely understood, certain factors such as copper and phosphorus content have been found to be significant, and the rules should encourage the trend toward use of radiation-insensitive materials. Furthermore, the definition of "beltline" was actually tightened to include all of the vessel that was expected to suffer more than 50°F

Charpy shift (rather than 100°F). This was to reduce the possibility of overlooking significant radiation damage to a shell course or a set of nozzles located just above or below the core. Thus, if radiation-sensitive materials are used in those locations, surveillance specimens must be taken from those materials as well as those directly opposite the core.

E. Objection to Prohibition on Direct Attachment of Radiation Capsules to the Vessel Wall

The weld attachment of capsule holders to the vessel wall should be permitted, because this minimizes fluence and temperature differences and causes no structural problems. (Combustion Engineering, General Electric).

Staff Action

The prohibition of direct attachment to the vessel wall remains in effect, although the merits of direct attachment are certainly recognized. The overriding concern was the possible introduction of a flaw in the beltline region of the vessel through improper weld practices, which might be used inadvertently praticularly if someone other than the vessel manufacturer did the welding.

Dated July 11, 1973, to become effective July 20, 1973.

CHARLES R. BRADER,
Acting Deputy Director, Fruit and Vegetable Division, Agricultural Marketing Service.

[FR Doc.73-14487 Filed 7-16-73;8:45 am]

CHAPTER X—AGRICULTURAL MARKETING SERVICE (MARKETING AGREEMENTS AND ORDERS; MILK), DEPARTMENT OF AGRICULTURE

[Milk Order No. 63]

PART 1063—MILK IN THE QUAD CITIES-DUBUQUE MARKETING AREA

Order Suspending Certain Provisions

This suspension order is issued pursuant to the provisions of the Agricultural Marketing Agreement Act of 1937, as amended (7 U.S.C. 601 et seq.), and of the order regulating the handling of milk in the Quad Cities-Dubuque marketing area.

Notice of proposed rulemaking was published in the FEDERAL REGISTER (38 FR 16878) concerning a proposed suspension of certain provisions of the order. Interested persons were afforded opportunity to file written data, views, and arguments thereon.

After consideration of all relevant material, including the proposal set forth in the aforesaid notice, data, views, and arguments filed thereon, and other available information, it is hereby found and determined that for the months of July and August 1973, the following provisions of the order do not tend to effectuate the declared policy of the Act:

In § 1063.14, the proviso which reads: *"Provided,* That in any of the months of July through January milk diverted from the farm of a producer on more than the number of days that milk was delivered to a pool plant from such farm during the month shall not be deemed to have been received by the diverting handler."

Statement of consideration. The suspension action will permit unlimited diversion of producer milk under the Quad Cities-Dubuque order during July and August 1973.

The suspension is requested by Mississippi Valley Milk Producers Association, Inc., Land O'Lakes, Inc., and Mid American Dairymen, Inc., to accommodate the handling of reserve milk on the market. The seasonal increase in milk production in conjunction with a decline in Class I sales has created a surplus milk disposal problem in this market. Without the suspension much of the reserve milk on the market would have to be moved from farms to pool plants and then reshipped to manufacturing plants in order to remain pooled instead of being moved directly from farms to manufacturing plants. The additional labor and hauling costs involved with such milk movements would adversely affect the economic handling of milk in excess of fluid requirements. This suspension will allow more economic handling of the market's reserve milk while the dairy farmers involved, retain producer status.

It is hereby found and determined that thirty days' notice of the effective date hereof is impractical, unnecessary and contrary to the public interest in that:

(a) This suspension is necessary to reflect current marketing conditions and to maintain orderly marketing conditions in the marketing area in that the most efficient method of handling the market's reserve milk supply is movement directly from producers' farms to milk manufacturing plants. This suspension would allow such handling during July and August 1973, while the dairy farmers involved retain producer status;

(b) This suspension order does not require of persons affected substantial or extensive preparation prior to the effective date; and

(c) Notice of proposed rulemaking was given interested parties and they were afforded opportunity to file written data, views or arguments concerning this suspension. No views were received in opposition to the proposed suspension.

Therefore, good cause exists for making this order effective with respect to producer milk deliveries during July and August 1973.

It is therefore ordered, That the aforesaid provisions of the order are hereby suspended for the months of July and August 1973.

(Secs. 1-19, 48 Stat. 31, as amended; 7 U.S.C. 601-674)

Effective date: July 17, 1973.

Signed at Washington, D.C., on July 12, 1973.

CLAYTON YEUTTER,
Assistant Secretary.

[FR Doc.73-14582 Filed 7-16-73;8:45 am]

Title 9—Animals and Animal Products

CHAPTER I—ANIMAL AND PLANT HEALTH INSPECTION SERVICE, DEPARTMENT OF AGRICULTURE

SUBCHAPTER C—INTERSTATE TRANSPORTATION OF ANIMALS (INCLUDING POULTRY) AND ANIMAL PRODUCTS; EXTRAORDINARY EMERGENCY REGULATION OF INTRASTATE ACTIVITIES

PART 72—TEXAS (SPLENETIC) FEVER IN CATTLE

Permitted Dips

The purpose of this amendment is to lower the concentration at which approved proprietary brands of Dioxathion (Delnav®) may be used as a permitted dip in official dipping for interstate movement.

Statement of consideration. The Environmental Protection Agency has recommended a change in the proposed use pattern for Dioxathion (Delnav®) when used on beef cattle, horses, sheep and goats which would reduce the maximum concentration at which this product may be used from 0.160 percent to 0.150 percent. Such a concentration is within the effective range for disinfection purposes.

Therefore, to conform with the Environmental Protection Agency's proposed use pattern and pursuant to the provisions of the Act of March 3, 1905, as amended, the Act of February 2, 1903, as amended, and the Act of May 29, 1884, as amended (21 U.S.C. 111-113, 115, 117, 120, 121, 123-126), § 72.13(b)(2) is amended to read as follows:

§ 72.13 Permitted dips and procedures.

* * * * *

(b) * * *

(2) Approved proprietary brands of a Dioxathion (Delnav®) emulsifiable concentrate used at a concentration of 0.125 to 0.150 percent.[3]

* * * * *

(Secs. 1, 2, 32 Stat. 791-792, as amended; secs. 4-7, 23 Stat. 32, as amended; secs. 1-4, 33 Stat. 1264, 1265, secs. 3 and 11, 76 Stat. 130, 132 as amended; 21 U.S.C. 111-113, 115, 117, 120, 121, 123-126, 134b, 134f; 37 FR 28464, 28477.)

Effective date. The foregoing amendment shall become effective July 17, 1973.

The amendment relieves certain restrictions presently imposed but no longer deemed necessary to prevent the interstate spread of Texas fever ticks and must be made effective promptly to be of maximum benefit to persons subject to the restrictions which are relieved. It does not appear that public participation in this rulemaking proceeding would make additional relevant information available to the Department.

Accordingly, under the administrative procedures provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and contrary to the public interest, and good cause is found for making the amendment effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 11th day of July 1973.

F. J. MULHERN,
Administrator, Animal and Plant Health Inspection Service.

[FR Doc.73-14581 Filed 7-16-73;8:45 am]

Title 10—Atomic Energy

CHAPTER I—ATOMIC ENERGY COMMISSION

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Fracture Toughness and Surveillance Program Requirements

On July 3, 1971 the Atomic Energy Commission published in the FEDERAL REGISTER (36 FR 12697) proposed amendments to its regulations in 10 CFR Part 50 which would add new appendices entitled, "Appendix G, Fracture Toughness Requirements," and "Appendix H, Re-

[3] Care is required when treating animals and in maintaining required concentration of chemicals in dipping baths. Detailed information concerning the use of, criteria for, and names of proprietary brands of permitted dips for which specific permission has been granted, and concerning the use of compressed air, vat management techniques, and vatside tests, and other pertinent information may be obtained from the U.S. Department of Agriculture, APHIS, Veterinary Services, Hyattsville, Maryland 20782.

actor Vessel Material Surveillance Program Requirements."

Interested persons were invited to submit written comments within 60 days. Upon consideration of the comments received and other factors involved, the Commission has adopted the proposed amendments with certain modifications in the form set forth below.

Significant differences in Appendix G from the amendments published for comment are:

(1) Terminology was changed to be consistent with that of the ASME Code.[1]

(2) The method of combining the results of the Charpy and dropweight tests to get a combined measure of toughness was changed.

The proposed rule would have required characterization of the fracture toughness of the ferritic materials in the reactor coolant pressure boundary in terms of the temperature dependence of two quantities: (a) Energy absorbed in Charpy V-notch impact tests (ASTM[2] Standard A-370) and (b) the nil-ductility transition (NDT) temperature obtained from dropweight tests (ASTM Standard E-208). Charpy tests were to be run at appropriate temperatures to characterize the transition from fully ductile, "upper shelf," behavior to low-energy, "brittle," behavior. To obtain a toughness characterization that depended on both types of tests, the "Charpy curve" was to be adjusted upward on the temperature scale to make the 15 ft. lb. level correspond to the NDT temperature from the dropweight tests.

These amendments continue the requirement contained in the proposed rule that fracture toughness be measured by the Charpy test and the dropweight test. However, to reflect comments urging consistency with the ASME Code, fracture toughness of the material is characterized by its reference temperature, RT_{NDT}. This temperature is the higher value of the NDT temperature from the dropweight test or the temperature that is 60° F below the temperature at which Charpy test data meet a specified toughness level (50 ft. lbs. and 35 mils lateral expansion).

(3) The concept of a lowest pressurization temperature given in the proposed rule was changed to a concept based on fracture mechanics that allows a continuous buildup of pressure as a function of temperature and wall thickness.

The proposed rule would have required a "thickness correction" whereby the Charpy curve was to be shifted up the temperature scale 7°F per inch of material thickness. The thickness correction would have been added to the shift required for consistency between the two types of toughness tests to obtain a curve of "adjusted fracture energy" versus temperature. Fracture control would have been achieved by requiring the "lowest pressurization temperature" at which system pressure could exceed 25 percent of normal operating pressure, or at which the rate of temperature change could exceed 50°F/hr., to be the temperature at which the adjusted fracture energy exceeded a certain level, which was higher for thick material than for thin.

Many of the comments questioned the validity of the dependence placed on the Charpy test by the proposed rule. The thickness correction was considered excessive for thick sections and inadequate for thin sections. Other comments asked that the rules treat stresses more quantitatively to take account of the operators' ability to control pressure and rate of temperature change and the designers' ability to calculate pressure and thermal stresses. Specifically, they urged the adoption of the approach that now appears in the 1972 Summer Addenda to the ASME Code. The proposed rules were also revised to reflect these comments. As required by these amendments, fracture control is achieved by requiring that stress in the pressure boundary be limited as a function of the metal temperature relative to the reference temperature, RT_{NDT}, and as a function of material thickness according to the "K_{IR} curve" given in the ASME Code. Taken from fracture mechanics, the term "stress intensity factor" (K) defines a quantity that is proportional to the product of gross stress and the square root of crack depth, and includes factors to account for crack shape and for the manner of loading. Critical values of K, determined from tests in which precracked specimens are loaded to failure, are a convenient measure of fracture toughness, because differences in crack size and shape and differences in manner of loading between specimen and component can be treated quantitatively. The K_{IR} curve in the ASME Code gives allowable values of fracture toughness as a function of temperature relative to RT_{NDT}. The curve is based on data obtained from tests of large specimens in the HSST[3] program. Rather than require the estimation of maximum expected flaw size, these amendments require that in areas of the reactor vessel remote from discontinuities, the assumed flaw size be proportional to wall thickness. Thus, from the value of K_{IR} at a given temperature, allowable stress values are obtained that are inversely proportional to the square root of wall thickness.

(4) Fracture control procedures described in paragraph (3), above, are supplemented in these amendments by a requirement that whenever the core is critical, the metal temperature of the reactor vessel shall exceed specified values dependent on the concurrent stress level.

(5) The Charpy V-notch upper-shelf energy requirements for beltline region materials was set at 75 ft. lbs. for all cases, without distinction as to the predicted amount if irradiation damage.

(6) Fracture toughness requirements for the various components of the pressure boundary were separated to reflect comments suggesting that the rules fit the anticipated severity of service to which the component might be subjected.

(7) The definition of "beltline region of the reactor vessel" was broadened to include more shell material above and below the core.

Significant differences in Appendix H from the amendments published for comment are:

(1) Terminology was changed to be consistent with that of Appendix G and the ASME Code. In particular, the adjustment for irradiation effects is described in these amendments as an adjustment of the reference temperature, RT_{NDT}, and the amount of temperature shift is determined by a slightly different treatment of the Charpy data than that given in the proposed amendment.

(2) Provision was made for accelerated irradiation capsules and for modification of capsule withdrawn schedules based on results of tests of specimens that received the accelerated irradiation.

(3) A general provision for an integrated surveillance program was substituted for the specific requirements given in the proposed rule. It appeared from comments that it would be impractical to meet the requirements of the proposed rule for commonality of multiple reactors.

Appendices G and H are intended to implement General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," to the extent described below. The margin of safety against brittle fracture will be controlled more quantitatively by these amendments than by the proposed rule, particularly with regard to specific guidelines for the treatment of heatup and cooldown conditions. Appendices G and H track the language of the ASME Code and have adopted certain of its requirements but also include several key supplemental requirements. For the vessel beltline, inservice requirements are based on the reference temperature as adjusted to account for irradiation damage. There is also an additional fracture toughness requirement in the form of shelf energy values from the Charpy curve for the material in its unirradiated condition.

Although the requirements of Appendices G and H become effective on August 16, 1973, the Commission recognizes that there may be an interim period when, for plants now under construction, the method of compliance with certain provisions may be determined on a case-by-case basis. For

[1] American Society of Mechanical Engineers Boiler and Pressure Vessel Code, section III, "Rules for the Construction of Nuclear Power Plant Components," 1971 Edition, and addenda through the Winter, 1972 Addenda.

[2] American Society for Testing and Materials.

[3] Heavy Section Steel Technology Program, conducted at Oak Ridge National Laboratory.

example, if the test data needed to establish certain fracture control requirements are not available because they were not required at the time material sampling was done, estimated values that are appropriately conservative may be acceptable.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Part 50, are published as a document subject to codification to be effective on August 16, 1973.

1. In § 50.55a of 10 CFR Part 50, the existing paragraph (i) is redesignated paragraph (j), a new paragraph (i) is added, and subdivision (a)(2)(i) and the prefatory language in paragraph (a)(2) are amended to read as follows:

§ 50.55a Codes and standards.

Each construction permit for a utilization facility shall be subject to the following conditions, in addition to those specified in § 50.55:

(a)(1) * * *

(2) As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (i) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing, or inspection of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety; or

* * * * *

(i) Fracture toughness requirements: Pressure-retaining components of the reactor coolant pressure boundary shall meet the requirements set forth in Appendices G and H to this part.

(j) Power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit has been published on or before December 31, 1970, may meet the requirements of paragraphs (c)(1), (d)(1), (e)(1), and (f)(1) of this section instead of paragraphs (c)(2), (d)(2), (e)(2), and (f)(2) of this section, respectively.

2. New Appendices G and H are added to Part 50 to read as follows:

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 drop-weight 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.

APPENDIX H—REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I. INTRODUCTION

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water cooled power reactors resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

II. SURVEILLANCE PROGRAM CRITERIA

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods, applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E 1MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm².

B. Reactor vessels constructed of ferritic materials which do not meet the conditions of section II.A. shall have their beltline regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, ASTM Designation: E-185-73,[1] except as modified by this appendix.

C. The surveillance program shall meet the following requirements:

1. Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by section III of Appendix G. The specimen types shall comply with the requirements of section III.A. of Appendix G (except that drop weight specimens are not required).

2. Surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region, so that the neutron flux received by the specimens is at least as high but not more than three times as high

[1] Effective March 1, 1973. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103, either as a separate or (when available) as part of the 1973 Annual Book of ASTM Standards, Part 30 and also in Part 31. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. NW., Washington, D.C.

as that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules, for which the calculated neutron flux will exceed three times the calculated maximum neutron flux at the inside wall of the vessel, may be used in addition to the required number of surveillance capsules specified in section II.C.3.

3. The required number of surveillance capsules and their withdrawal schedules are as follows:

a. For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steel, making appropriate allowances for all uncertainties in the measurements, that the adjusted reference temperature established in accordance with section IV.B. will not exceed 100°F at the end of the service lifetime of the reactor vessel, at least three surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule—One-fourth service life
Second capsule—Three-fourths service life
Third capsule—Standby

In the event that the surveillance specimens exhibit, at one-quarter of the vessel's service life, a shift of the reference temperature greater than originally predicted for similar material as recorded in the applicable technical specification, the remaining withdrawal schedule shall be modified as follows:

REVISED
WITHDRAWAL SCHEDULE

Second capsule—One-half service life
Third capsule—Standby

b. For reactor vessels which do not meet the conditions of section II.C.3.a. but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted reference temperature will not exceed 200°F at the end of the service lifetime of the reactor vessel, at least four surveillance capsules shall be provided for the subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule—At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.
Second capsule—At approximately one-half of the time interval between first and third capsule withdrawal.
Third capsule—Three-fourths service life.
Fourth capsule—Standby.

c. For reactor vessels which do not meet the conditions of section II.C.3.b., at least five surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule—At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.
Second and third capsules—At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.
Fourth capsule—Three-fourths of service life.
Fifth capsule—Standby.

d. Provision shall also be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation.

e. Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

f. If accelerated irradiation capsules are employed in addition to the minimum required number of surveillance capsules, the withdrawal schedule may be modified, taking into account the test results obtained from testing of the specimens in the accelerated capsules. The proposed modified withdrawal schedule in such cases shall be approved by the Commission on an individual case basis.

g. Proposed withdrawal schedules that differ from those specified in paragraphs a. through f. shall be submitted, with a technical justification therefor, to the Commission for approval. The proposed schedule shall not be implemented without prior Commission approval.

4. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

III. FRACTURE TOUGHNESS TESTS

A. Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of section III of Appendix G, "Fracture Toughness Requirements."

B. The adjusted reference temperatures for the base metal, heat-affected zone, and weld metal shall be obtained from the test results by adding to the reference temperature the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 foot-pound level or that measured at the 35 mil lateral expansion level, whichever temperature shift is greater. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements of section V.B. of Appendix G are satisfied.

IV. REPORT OF TEST RESULTS

A. Each capsule withdrawal and the results of the fracture toughness tests shall be the subject of a summary technical report to be provided to the Commission. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the relationship of the measured results to those predicted for the reactor vessel beltline region.

B. The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values of fluence.

C. The operating pressure and temperature limitations established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes made in operational procedures to assure meeting such temperature limitations.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Germantown, Md., this 11th day of July 1973.

For the Atomic Energy Commission.

GORDON M. GRANT,
Acting Secretary of the Commission.

NOTE: Incorporation by reference provisions approved by the Director of the Federal Register on May 29, 1973.

[FR Doc.73-14531 Filed 7-16-73;8:45 am]

Title 12—Banks and Banking

CHAPTER II—FEDERAL RESERVE SYSTEM

SUBCHAPTER A—BOARD OF GOVERNORS OF THE FEDERAL RESERVE SYSTEM

PART 201—EXTENSIONS OF CREDIT BY FEDERAL RESERVE BANKS

Changes in Rates

Pursuant to section 14(d) of the Federal Reserve Act (12 U.S.C. 357), and for the purpose of adjusting discount rates with a view to accommodating commerce and business in accordance with other related rates and the general credit situation of the country, Part 201 is amended as set forth below:

1. Section 201.51 is amended to read as follows:

§ 201.51 Advances and discounts for member banks under sections 13 and 13a.

The rates for all advances and discounts under sections 13 and 13a of the Federal Reserve Act (except advances under the last paragraph of such section 13 to individuals, partnerships, or corporations other than member banks) are:

Federal Reserve Bank of	Rate	Effective
Boston	7	July 2, 1973
New York	7	July 2, 1973
Philadelphia	7	July 2, 1973
Cleveland	7	July 2, 1973
Richmond	7	July 2, 1973
Atlanta	7	July 2, 1973
Chicago	7	July 2, 1973
St. Louis	7	July 2, 1973
Minneapolis	7	July 2, 1973
Kansas City	7	July 2, 1973
Dallas	7	July 2, 1973
San Francisco	7	July 2, 1973

2. Section 201.52 is amended to read as follows:

§ 201.52 Advances to member banks under section 10(b).

The rates for advances to member banks under section 10(b) of the Federal Reserve Act are:

Federal Reserve Bank of	Rate	Effective
Boston	7-1/4	July 2, 1973
New York	7-1/4	July 2, 1973
Philadelphia	7-1/4	July 2, 1973
Cleveland	7-1/4	July 2, 1973
Richmond	7-1/4	July 2, 1973
Atlanta	7-1/4	July 2, 1973
Chicago	7-1/4	July 2, 1973
St. Louis	7-1/4	July 2, 1973
Minneapolis	7-1/4	July 2, 1973
Kansas City	7-1/4	July 2, 1973
Dallas	7-1/4	July 2, 1973
San Francisco	7-1/4	July 2, 1973

3. Section 201.53 is amended to read as follows:

§ 201.53 Advances to persons other than member banks.

The rates for advances under the last paragraph of section 13 of the Federal Reserve Act to individuals, partnerships, or corporations other than member banks secured by direct obligations of, or obligations fully guaranteed as to principal and interest by, the United States or any agency thereof are:

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Domestic Licensing of Production and Utilization Facilities; Fracture Toughness Requirements for Nuclear Power Reactors

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is considering amending its regulations which specify fracture toughness requirements for nuclear power reactors and its requirements for reactor vessel material surveillance programs. The amendments would clarify the applicability of these requirements to old and new plants, modify certain requirements as described below, and shorten and simplify these regulations by more extensively incorporating by reference appropriate National Standards.

DATES: Comment period expires January 13, 1981.

ADDRESSES: Written comments should be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

FOR FURTHER INFORMATION CONTACT: Dr. P. N. Randall, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, 301-443-5997.

SUPPLEMENTARY INFORMATION: When 10 CFR Part 50 was amended by adding Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," on July 17, 1973 (38 FR 19012), the ASME Boiler and Pressure Vessel Code (the ASME Code) provisions pertaining to nuclear power plant components were being extensively revised. Some requirements that had not yet become effective in the ASME Code were included in Appendix G. Now that experience has shown that the ASME Code requirements are adequate, Appendix G is being condensed by more extensively incorporating the ASME Code by reference.

Appendix H incorporated by reference the 1973 edition of ASTM E 185.[1] Now there is a new edition, E 185-79. As amended, Appendix H would specify the earliest edition of E 185 that could be used for each part of the surveillance program, and would delete paragraphs that are now covered by specific provisions of E 185-79.

Paragraph 50.55a(i), which makes the provisions of Appendices G and H conditions of a construction permit for a utilization facility (e.g., a nuclear power plant), would be deleted and a new § 50.47 would be added for this purpose. This change would avoid any confusion that might occur when § 50.55a is amended for purposes not relevant to the appendices. The new § 50.47 also would describe when an exemption must be requested of the Commission for acceptance of a proposed alternative to the requirements of Appendices G and H. Conversely, several paragraphs of Appendices G and H would be amended to state that a proposed alternative may be accepted by the Director of Nuclear Reactor Regulation if the proposed alternative provides a margin of safety equivalent to that achieved by the specified requirement.

Significant differences between the existing regulations in Appendix G and the proposed amendments are:[2]

¶II.G. The definition of "adjusted reference temperature" would be changed to make the definition applicable to test data over a broader range of severity of radiation damage.

¶II.H. The definition of "beltline" would be changed to more clearly delineate which parts of the vessel are of concern from the standpoint of surveillance of radiation damage.

¶II.J. The definition of "integrated surveillance program" would be removed from Appendix G and a more complete discussion of such a program would be included in paragraph II.C. of Appendix H.

§ III. Section III, "Fracture Toughness Tests," would be reduced in length because most of the pertinent requirements could be incorporated by referencing the ASME Code.

¶III.A. Language would be added to paragraph III.A. to clarify how the fracture toughness test requirements would be applied to "old" plants, those for which the reactor vessel was constructed to an ASME Code earlier than the Summer 1972 Addenda to the 1971 Edition. This language would clarify the Commission's intention that all plants must meet the required margins of safety and the other fracture prevention requirements of Appendices G and H; but the owners of "old" plants may, when approved by the Director of Nuclear Reactor Regulation, use fracture toughness tests and fracture analyses other than those required by Appendices G and H, to demonstrate that the required margins of safety have been met.

§ IV. As amended, Section IV, "Fracture Toughness Requirements" would incorporate by reference the fracture toughness requirements of the ASME Code and then give certain supplemental requirements, thereby deleting much technical detail from the regulation.

¶IV.A.2.b. As an alternative to the existing general requirement for fracture toughness of nozzles, flanges and shell regions near structural discontinuities, a specific pressure-temperature requirement would be added.

¶IV.A.2.c. For boiling water reactors, the criticality limit would be reduced to permit them to go critical earlier during startup. A systems analysis has shown that the existing limit is more restrictive than is necessary to provide the required safety margin.

¶IV.A.2.d. The minimum permissible temperature for system hydrostatic pressure tests would be reduced when performed when there is no fuel in the reactor (and thus no radiological hazard to the public) to improve the efficiency of the inspection operation.

¶IV.B. The 50 ft lb requirement for Charpy upper-shelf energy would be stated explicitly. Language would be added to permit acceptance of a lower value by the Director of Nuclear Reactor Regulation if a fracture analysis showed that the margin of safety against ductile fracture was equivalent to the margin of safety against fracture in the transition region required by Appendix G of the ASME Code.

¶V.B. A requirement would be added that material toughness values to be used in fracture analyses are those predicted for the material near the tip of the assumed flaw at its deepest part.

Significant differences between the existing regulations in Apendix H and the proposed amendments are:

¶II.B. As revised, paragraph II.B. would incorporate by reference ASTM E 185, and reference to the 1973 edition of ASTM E 185 would be deleted. Language would be added to permit the use of the 1979 edition or an earlier edition if it was the current edition at the time the action was taken. For example, the earliest

[1] Standard Recommended Practice for Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H St., NW., Washington, D.C.

[2] Listed according to existing paragraph numbers. See Enclosure 3.

edition of ASTM E 185 that could be used in the selection of surveillance materials, preparation of specimens, and construction of surveillance capsules would be the edition that was current on the issue date of the ASME Code to which the reactor vessel was purchased.

¶II.C.3. Most of this paragraph would be deleted, because the requirements for withdrawal schedules contained in the 1979 edition of ASTM E 185 provide satisfactory criteria for scheduling surveillance information gathering.

¶II.C.4. Expanded criteria would be given for an integrated surveillance program for a set of reactors of similar design and operating features. In an integrated program, surveillance capsules from one or more reactors are irradiated in other reactors in the set.

§III. Section III would be deleted because its provisions are covered in Appendix G, Section V.

§IV. Reporting requirements for the test results from each capsule withdrawal would be clarified and a time limit for submittal of the report would be specified.

Staff of the Commission's Office of Standards Development has prepared a value/impact statement for the proposed amendment, which provides additional technical details and justification. This statement is available for inspection by the public in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C. Single copies of the value/impact statement may be obtained by request addressed to the Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: P. N. Randall, (301) 443-5997.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganizaiton Act of 1974, as amended, and section 553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendments of 10 CFR Part 50 is contemplated. All interested persons who wish to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch by Janaury 13, 1981. Copies of comments received may be examined in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C.

1. Section 50.55a is amended by deleting paragraph (i) and inserting the word "Reserved" in its place.

2. A new § 50.47 is added to 10 CFR Part 50 to read as follows:

§ 50.47 Acceptance criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors.

(a)(1) Except as provided in paragraph (a)(2) of this section all light-water nuclear power reactors shall meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.

(2) Proposed alternatives to the described requirements or portions therof may be used when an exemption is granted by the Commission as authorized by § 50.12. In addition, the applicant must demonstrate that (i) compliance with the specified requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety and (ii) the proposed alternatives would provide an adequate level of quality and safety.

(b) [Reserved]

* * * * *

§ 50.12 [Amended]

3. Paragraph (a) of § 50.12 is amended by adding the following sentence at the end of the paragraph:

* * * To obtain an exemption to Appendices G and H to this part, the requirements of paragraph 50.47(a)(2) must be met in addition to the requirements of this paragraph.

4. Appendices G and H to 10 CFR Part 50 are revised to read as follows:

Appendix G—Fracture Toughness Requirements

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I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear 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 (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G-2110 of the ASME Code.[1]

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 (896 MPa).

Adequacy of the fracture toughness of other ferritic materials shall be demonstrated to the Director of Nuclear Reactor Regulation on an individual case basis.

II. Definitions

A. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified, the applicable ASME Code edition and addenda [2] and any limitations and modifications thereof are specified by § 50.55a, Codes and Standards.

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 crystal 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.

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. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in the ASME Code.

F. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Section V of this Appendix) by adding to RT_{NDT} the temperature shift in the average Charpy curve for the irradiated material relative to that for the unirradiated material, measured at the 30 ft lb (41J) level.

G. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

[1] Defined in paragraph II.A. The latest Edition and Addenda permitted by paragraph 50.55a(b) at the time the analysis is made shall be used for the purposes of paragraph I.A.

[2] 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.

III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Sections IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (pursuant to § 50.55a), the fracture toughness data and data analyses shall be supplemented in a manner approved by the Director of Nuclear Reactor Regulation to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph V.C.2 shall be submitted to and approved by the Director of Nuclear Reactor Regulation prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs A and B of this Section shall comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

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

1. Reactor vessel beltline materials shall have minimum Charpy upper-shelf-energy of 75 ft lb (102J) initially and shall maintain minimum upper-shelf energy throughout the life of the vessel of 50 ft lb (68J), unless it is demonstrated in a manner approved by the Director of Nuclear Reactor Regulation that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.[3]

2. When the core is not critical, pressure-temperature limits for the reactor vessel shall be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code [3] supplemented by the requirements of Section V of this Appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the stressed regions of nozzles, flanges and other structural discontinuities shall be at least 150° F above (83 C above) the reference temperature of the material in those regions unless a lower temperature can be justified by showing that the margins of safety for those regions are equivalent to those required for the beltline when it is controlling. The justification submitted for the pressure temperature limits shall describe the methods of analysis used.

3. When the core is critical (other than for the purpose of low-level physics tests) the temperature of the reactor vessel shall not be lower than 40° F above (22 C above) the minimum permissible temperature of paragraph 2. of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure, in which case the minimum permissible temperature is 60° F above (33 C above) the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

4. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature shall be $RT_{NDT}+60°$ F ($RT_{NDT}+33$ C).

5. If there is fuel in the reactor during system pressure tests or leak tests, the requirements of paragraphs 2 or 3 shall apply, depending on whether the core is critical during the test.

B. Reactor vessels for which the predicted value of upper shelf-shelf energy at end of life is below 50 ft lb or the predicted value of adjusted reference temperature at end of life exceeds 200° F (93 C) 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 effects of neutron radiation on the reference temperature and upper-shelf energy of reactor vessel beltline materials, including welds, shall be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions shall be made for the radiation conditions at the tip of the assumed flaw at its deepest part. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements are satisfied.

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. A volumetric examination of the beltline materials that do not satisfy the requirements of Section V.B. including 100 percent of any associated welds shall be made and any flaws evaluated according to Section XI of the ASME Code and as otherwise specified by the Director of Nuclear Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation shall be obtained from results of supplemental fracture toughness tests.

3. An analysis shall be performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

D. If the procedures of section V.C. do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may, subject to the approval of the Director of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material. The degree of recovery shall be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-at-temperature conditions as those given the beltline material. The results shall provide the basis for establishing the adjusted reference temperature and upper shelf energy 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. using the values of adjusted reference temperature and upper-shelf energy 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 Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, 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.

Appendix H—Reactor Vessel Material Surveillance Program Requirements

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I. Introduction

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Sections IV and V of Appendix G.

II. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E<1MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

[3] The latest Edition and Addenda permitted by paragraph 50.55a(b) at the time the analysis is made shall be used for the purposes of paragraphs IV.A.1 and IV.A.2.

B. Reactor vessels that do not meet the conditions of paragraph II.A. shall have their beltline materials monitored by a surveillance program complying with ASTM E 185,[1] except as modified by this Appendix. ASTM E 185 was approved for incorporation by reference by the Director of the Federal Register, on May 29, 1973.

1. That part of the surveillance program conducted prior to the first capsule withdrawal shall meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. For each capsule withdrawal, the test procedures and reporting requirements shall meet the requirements of the edition of E 185 in effect on the date of capsule withdrawal, to the extent practical for the configuration of the specimens in the capsule. For any part of the surveillance program, later editions of E 185 may be used instead of the editions previously specified, but including only those editions through 1979.

2. Surveillance specimen capsules shall be located near the inside vessel wall in the beltline region, so that the specimen irradiation history duplicates to the extent practicable, within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds shall be done according to the requirements for permanent structural attachments to reactor vessels given in the ASME Code, Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

3. Proposed withdrawal schedules shall be submitted with a technical justification therefor to the Director of Nuclear Reactor Regulation for approval. The proposed schedule shall not be implemented without prior approval.

C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance programs must be approved by the Director of Nuclear Reactor Regulation on a case-by-case basis. Criteria for approval include the following considerations:

1. The design and operating features of the reactors in the set shall be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

2. There shall be adequate arrangements for data sharing between plants.

3. There shall be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

4. There shall be substantial advantages to be gained in reduced power outages or personnel exposure to radiation, achieved by not requiring surveillance capsules in all reactors in the set.

III. Report of Test Results

A. Each capsule withdrawal and the results of the fracture toughness tests shall be the subject of a summary technical report to be submitted for approval to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 90 days after completion of testing. The Director shall be notified at least 30 days in advance of the capsule withdrawal, giving the expected date of completion of testing and submittal of report.

B. The report shall include the data required by ASTM E 185, as required by paragraph II.B.1, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the operating pressure-temperature limits given in the Technical Specifications is required, the revised limits shall be submitted with the report, including any changes made in operating procedures required to meet the limits.

* * * * *

(Secs. 103, 104, 161i, Pub. L. 83-703; 68 Stat. 936, 937, 948; Sec. 201, Pub. L. 93-438, 88 Stat. 1242; (42 U.S.C. 2133, 2134, 2201(i), 5841))

Dated at Washington, D.C. this 5th day of November 1980.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 80-35307 Filed 11-13-80; 8:45 am]

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[1] Standard Recommended Practice for Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H St., NW., Washington, D.C.