UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY	)	Docket No. 50-440-OLA-3
	)	
	)	
(Perry Nuclear Power Plant, Unit 1)	)	(Material Withdrawal Schedule)
	)	

AFFIDAVIT

Barry J. Elliot (BJE), Jack R. Strosnider (JRS) and Christopher I. Grimes (CIG), being first duly sworn, do dispose and state as follows:

1(a). (BJE) I am employed by the U.S. Nuclear Regulatory Commission as a Senior Materials Engineer in the Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached hereto.

1(b). (JRS) I am employed by the U.S. Nuclear Regulatory Commission as Chief of the Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached hereto.

1(c). (CIG) I am employed by the U.S. Nuclear Regulatory Commission as Chief of the Technical Specifications Branch, Division of Operating Reactor Support, Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached hereto.

2. (BJE, JRS, CIG) The purpose of this affidavit is (a) to explain the reasons for NRC Staff's determination that a licensee's nuclear reactor vessel material specimen capsule

withdrawal schedule may be removed from the licensee's technical specifications (TS), as previously provided in Generic Letter (GL) 91-01, entitled "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," dated January 4, 1991, (b) to describe the Staff's practice in reviewing requests for approval of changes to a licensee's withdrawal schedule pursuant to 10 C.F.R. Part 50, Appendix H, and (c) to respond to the questions raised by the Licensing Board in its Order of December 27, 1993.

3. (BJE, JRS) The Materials and Chemical Engineering Branch, Division of Engineering, is responsible for approving withdrawal schedules submitted for review in accordance with 10 C.F.R. Part 50, Appendix H, entitled "Reactor Vessel Material Surveillance Program Requirements." As part of our duties, we reviewed and approved the relocation of the withdrawal schedule from the Perry Nuclear Power Plant's TS, to be inserted in the plant's Updated Safety Analysis Report (USAR), as had been requested by the licensee in its license amendment application dated March 15, 1991.

4. (BJE, JRS) Appendix H to 10 C.F.R. Part 50 provides a means for obtaining test data that can be used in monitoring the effects of neutron irradiation and the thermal environment on reactor vessel beltline materials. The Introduction to Appendix H states, in part:

> The purpose of the materials 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.

The importance of the material specimen surveillance program is discussed below.

5. (BJE, JRS) Paragraph II.B. of Appendix H provides, in pertinent part, as follows:

> 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.
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> * * *
>
> 3. A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4. The proposed schedule must be approved prior to implementation.[1]

Compliance with Appendix H is required by 10 C.F.R. § 50.60(a), although alternatives to those requirements may be proposed by a licensee pursuant to 10 C.F.R. §§ 50.60(b) and 50.12.

6. (BJE, JRS) While Appendix H (§ II.B.1.) is clear that a licensee's initial specimen withdrawal program must comply with the applicable edition of ASTM E 185, it does not explicitly address the requirements for changes to a previously approved withdrawal schedule,

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[1] The Introduction to Appendix H notes that "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."

and is ambiguous as to how such changes are to be reviewed and approved. In this regard, the regulatory history of Appendix H provides clarification. Previous iterations of Appendix H, prior to 1983, set forth specific withdrawal schedules which were required to be followed by NRC licensees. However, in 1983, the Commission issued an amendment to Appendix H (48 Fed. Reg. 24008), in which it deleted the withdrawal schedules, but retained the references to ASTM E 185 and the requirement that withdrawal schedules must be approved by the Staff prior to implementation. As indicated in explanatory documents (including the supplementary information for the proposed rule, the value/impact statement, the regulatory analysis, and responses to comments), prepared in conjunction with the changes to Appendix H, ASTM E 185-79 (the 1979 edition of the standard) contained sufficient detail for the preparation of withdrawal schedules to permit the deletion of withdrawal schedules from Appendix H. Accordingly, proposed withdrawal schedules or changes which were in conformance with ASTM E 185-79 (or ASTM E 185-82, which contains identical withdrawal schedule criteria) would satisfy the requirements of Appendix H. Subsequent to the rule change, the Staff reviewed proposed schedules and modifications to determine if they were consistent with the withdrawal schedules set forth in ASTM E 185 or were otherwise acceptable. This review was normally conducted as part of a license amendment proceeding, since matters located in a licensee's TS (such as the withdrawal schedules) could only be changed by license amendment as set forth in 10 C.F.R. § 50.59(c).

7. (CIG) Without focusing upon any particular TS requirements, the Commission has long expressed concern over the volume of TS requirements for nuclear power reactors. For example, in March 1982, the Commission issued a proposed rule change which would have

reduced the volume of technical specifications in operating licenses, indicating that such a change would constitute an improvement in the safety of nuclear plants through more efficient use of licensee and NRC resources, and would help to focus licensee attention on matters of more immediate importance to safe operation of their facilities (47 Fed. Reg. 13369). While adoption of the proposed rule change was later deferred, the Commission has continued to recognize the desirability of reducing the volume of technical specifications, as indicated in an interim policy statement issued in February 1987 (52 Fed. Reg. 3788) and a final policy statement issued in July 1993 (58 Fed. Reg. 39132).

8. (CIG) In accordance with the Commission's interim policy statement of February 1987, among the actions taken by the Staff was the development of a program to improve the technical specifications for nuclear power reactors on a line-item basis. Several potential line-item TS improvements were identified by the Staff and reviewed by the NRC's Committee to Review Generic Requirements (CRGR), and were then made available for voluntary implementation through the issuance of generic letters.

9. (BJE, JRS, CIG) In late 1990, as part of the line-item TS improvement program, the Staff determined that material specimen capsule withdrawal schedules need not be retained in a facility's TSs, consistent with the criteria in the Commission's interim policy statement. The Staff determined that inclusion of the withdrawal schedules in the TS (a) was not specifically required by 10 C.F.R. § 50.36 or other regulations, (b) was not required to avert an immediate threat to the public health and safety, and (c) was not necessary since Appendix H provides an adequate means of controlling proposed changes to withdrawal schedules.

10. (BJE, JRS, CIG) In this regard, the Staff determined that while 10 C.F.R. § 50.36(c) requires that the TS "include" items in five listed categories, including "surveillance requirements," it nowhere specifies the particular surveillance requirements to be included in a plant's TS, nor does it require the inclusion of the capsule withdrawal schedules in the TS. In addition, the Staff determined that as long as the schedules are available for reference in the USAR by licensees and other persons, inclusion of the withdrawal schedules in the TS is not required. Further, as noted above, the Staff determined that Appendix H already provides sufficient regulatory controls to ensure the appropriateness of a capsule withdrawal schedule.

11. (BJE, JRS, CIG) The "surveillance requirements" to be included in a facility's TS, under 10 C.F.R. § 50.36(c), are "requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met." The Staff concluded that 10 C.F.R. § 50.36 does not require inclusion of the withdrawal schedule in the TS because that schedule is not "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." (*See* "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," Section IV, 58 Fed. Reg. 39132, 39136 (1993)). Instead, the TS include limiting conditions for operation and surveillance requirements for the reactor coolant system pressure and temperature (P-T) limits; these are "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." Maintaining the reactor coolant system within the P-T limits, along with compliance with other requirements of the

regulations, will "assure . . . the necessary quality of systems and components" is maintained and that facility will be operated "within the safety limits."

12. (BJE, JRS, CIG) Accordingly, on January 4, 1991, the Staff issued Generic Letter (GL) 91-01, entitled "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications." Therein, the Staff indicated that § II.B.3. of Appendix H requires NRC approval of a proposed withdrawal schedule prior to implementation, that "placement of this schedule in the TS duplicates the controls on changes to this schedule that have been established by Appendix H," that "this duplication is unnecessary," and that "removal of this TS schedule as a line-item improvement is consistent with the Commission [Interim] Policy Statement on TS Improvements." In addition, the Staff indicated as follows:

> The current STS bases provide extensive background information on the use of the data obtained from material specimens. This background information clearly defines the purpose and relationship of this information to the requirements included in the regulations and the American Society of Mechanical Engineers (ASME) Code. Therefore, the removal of the schedule for specimen withdrawal from the TS will not result in any loss of clarity related to regulatory requirements of Appendix H to 10 CFR Part 50.

(GL 91-01, Enclosure at 1). The Staff indicated it would approve the removal of withdrawal schedules from the TS, subject to a requirement that licensees doing so commit to include the schedules in the next revision of their Updated Safety Analysis Reports (USARs), so as to make a copy of the schedule readily available for licensees, NRC personnel and others.

13. (BJE, JRS) On March 15, 1991, the Licensee for the Perry Nuclear Power Plant requested that the withdrawal schedule be removed from the Perry TS and relocated to the

plant's USAR. At that time, the Perry TS, § 4.4.6.1.3., had described this surveillance as follows:

> The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR 50, Appendix H *in accordance with the schedule in Table 4.4.6.1.3-1*. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1. (Emphasis added.)

On December 18, 1992, in response to the Licensee's license amendment application, the Staff modified this surveillance by removing the phrase "in accordance with the schedule in Table 4.4.6.1.3-1" (italicized above) and the referenced Table, noting that the amendment was consistent with GL 91-01; the other provisions of the TS remain unchanged.

14. (BJE, JRS) The removal of a licensee's withdrawal schedule from its TS, in accordance with GL 91-01, does not relieve the licensee from the requirements of 10 C.F.R. Part 50, Appendix H. As indicated above, Appendix H was amended in 1983 by removing the withdrawal schedules from the regulation; however, ASTM E 185 was incorporated by reference in Appendix H, and the Commission indicated its intent that licensee withdrawal schedules are to be consistent with the schedule criteria contained in ASTM E 185-79 or -82. After a licensee has removed its withdrawal schedule from its TS, it may proceed to make changes to its schedule which are consistent with ASTM E 185-79 or -82, without prior NRC approval, and report those changes in a manner consistent with 10 C.F.R. § 50.59; however, if a licensee proposes schedule changes that are not consistent with ASTM E 185-79 or -82, the changes would likely be deemed to involve an unreviewed safety question under the current regulatory framework and would require prior NRC approval by a license amendment as provided by 10 C.F.R. § 50.59(c).

15. (BJE, JRS, CIG) The Staff has undertaken to review the wording of GL 91-01, and recognizes that it does not provide a clear understanding of these matters. The Staff is developing a clarification of the statements contained in that document, consistent with the statements presented here, and will also consider whether rulemaking is necessary to make explicit in Appendix H the circumstances under which the changes to a previously approved withdrawal schedule can be made.

16. The following information is provided in response to the questions raised by the Licensing Board in its Order of December 27, 1993:

> Question a. What is the relationship, if any, of 10 C.F.R. § 50.36 to the petitioners' contention?

Response: (BJE, JRS, CIG) There is no apparent relationship. As stated above, the withdrawal schedule is not required to be set forth in the TS surveillance requirements in order to satisfy the provisions of 10 C.F.R. § 50.36.

> Question b. Under Part 50, Appendix H, II.B.1., are there any changes in the reactor vessel material surveillance program withdrawal schedule that would not be reflected in a change in the limiting conditions of operation of the Perry facility?

Response: (BJE, JRS) Yes. The specimen withdrawal schedule is part of a licensee's program for monitoring the radiation embrittlement of the reactor vessel. Specimen materials are tested after being withdrawn from the reactor vessel, and the results obtained in those tests are used to confirm the amount of embrittlement previously assumed in the pressure-temperature (P-T) limit curves for the reactor vessel. The P-T limits are among the limiting conditions for operation (LCOs) for the reactor vessel; P-T limit curves define the acceptable range of reactor vessel temperatures and pressures for different operating conditions. For example, the P-T limit curves contained in the Perry TS apply to the following operating conditions: (a) system hydro or leak testing limit with fuel in the vessel, (b) the non-nuclear heatup/cooldown and physics test limit, and (c) the nuclear (core critical) limit.

The material properties used by a licensee in its P-T calculations are those properties which are conservatively assumed to exist in the future upon exposure of the vessel to neutron irradiation and the thermal environment, expressed in a specified number of effective full power years (EFPY) of plant operation. Changes to the P-T limits need to be made upon the receipt of information which indicates that the reactor vessel's material properties assumed by the licensee in its prior determination of the reactor's P-T limit curves are less conservative than is appropriate. Thus, Appendix H, § III.C. ("Report of Test Results"), contemplates that specimen test results may necessitate a change in the TS for "the pressure-temperature limits or in the operating procedures required to meet the limits."

Changes to the specimen withdrawal schedule may be made for a variety of reasons, such as to be consistent with revisions to the ASTM standard, or in response to the detection of a significant change in a specimen's material properties. Changes to the LCO could result based upon test data obtained in the monitoring program, but such changes would not result merely because the withdrawal schedule has been changed. Where tests on surveillance materials indicate that the assumed material properties for the P-T limits remain applicable, changes to the withdrawal schedule would not require a change in P-T limits.

Question c. If, as posited in Generic Letter 91-01 (Jan. 4, 1991), the removal of the reactor vessel material surveillance program withdrawal schedule from a facility's technical specifications will not result in any loss of clarity related to the requirements of Part 50, Appendix H, how is the removal of this duplicative matter from a facility's technical specifications violative of 10 C.F.R. § 50.36?

Response: (BJE, JRS, CIG) As stated above, removal of the specimen withdrawal schedule from a facility's TS does not violate any provision of 10 C.F.R. § 50.36.

17. (BJE, JRS, CIG) The foregoing statements are true and correct to the best of our knowledge, information and belief.

Barry J. Elliot

Barry J. Elliot

Jack R. Strosnider

Jack R. Strosnider

Christopher I. Grimes

Christopher I. Grimes

Subscribed and sworn to before me
this 7th day of March, 1994

Melinda L. McDonald

Notary Public

My commission expires: 12/1/97

BARRY J. ELLIOT

MATERIALS AND CHEMICAL ENGINEERING BRANCH
DIVISION OF ENGINEERING
OFFICE OF NUCLEAR REACTOR REGULATION
U.S. NUCLEAR REGULATORY COMMISSION

STATEMENT OF PROFESSIONAL QUALIFICATIONS

I am currently employed by the U.S. Nuclear Regulatory Commission as a Senior Materials Engineer in the Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. I am responsible for the review and the evaluation of safety analysis reports which are related to the materials engineering aspects of components in nuclear power plant systems. I also provide technical assistance to the Office of Nuclear Reactor Regulation and Nuclear Regulatory Research on related reactor safety matters.

I have been employed at the Nuclear Regulatory Commission since March 1980. I served as a Materials Engineer in the Office of Nuclear Reactor Regulation from 1980 to 1988, and was promoted to Senior Materials Engineer in February 1988.

I graduated from Rensselaer Polytechnical Institute in 1968 with a Bachelor of Science degree in Materials Engineering. I later received a Masters of Science degree in Business Administration from Fairleigh Dickinson University in 1971.

I was employed by Curtiss Wright Corporation from 1968 to 1980. From 1968 to 1971, I worked in the Materials Development Laboratory of the company's Aeronautical Division, where I performed failure analyses on reciprocating and gas-turbine engines and developed test apparatus to evaluate material reliability. From 1971 to 1980, I worked in the company's Nuclear Division, where I was responsible for developing and implementing non-destructive examination test procedures and fusion weld procedures to be used in the fabrication and inspection of U.S. Department of the Navy nuclear pressure vessels.

JACK R. STROSNIDER, JR.

U.S. NUCLEAR REGULATORY COMMISSION
MATERIALS AND CHEMICAL ENGINEERING BRANCH
DIVISION OF ENGINEERING
OFFICE OF NUCLEAR REACTOR REGULATION

STATEMENT OF PROFESSIONAL QUALIFICATIONS

EXPERIENCE:

SEPT 1991 to PRESENT — CHIEF
MATERIALS AND CHEMICAL ENGINEERING BRANCH
U.S. NUCLEAR REGULATORY COMMISSION

In this position I am responsible for managing the technical and administrative activities related to materials and chemical engineering aspects of reactor safety. This includes the technical and safety review of applications for license amendments for operating reactors. I supervise twenty-two engineers. Specific technical areas for which I am responsible include reactor pressure vessel integrity, steam generator tube integrity, and inservice inspection programs.

AUG 1990 to JULY 1992 — ADMINISTRATOR
NUCLEAR SAFETY DIVISION
OECD NUCLEAR ENERGY AGENCY

In this position I organized and administered international programs directed at maintaining and improving the safety of commercial nuclear power facilities. I was responsible for international programs on reactor component integrity; nondestructive testing of reactor components; conduct of regulatory safety inspections; and a project to assess the margin-to-failure of the reactor pressure vessel during the TMI-2 accident. The programs consisted of international research projects, workshops and seminars, and exchange of information. This position required extensive interaction with senior representatives and technical experts from the participating countries.

OCT 1986 to JULY 1990 — CHIEF, MATERIALS AND PROCESSES SECTION
U.S. NUCLEAR REGULATORY COMMISSION
REGION I

In this position I planned and implemented the Region I program for inspecting licensee materials, structural and mechanical engineering programs. I supervised six engineers who performed audits of licensee piping, vessel, and steam generator inspections and structural, plant modification, welding, and other engineering activities. I was responsible for the NRC Mobile Nondestructive Testing Laboratory Program. This included supervising three qualified technicians who performed nondestructive examinations including radiography, ultrasonic, magnetic particle, dye penetrant and visual examinations in accordance with applicable codes and standards.

SEPT 1984 to OCT 1986 — CHIEF, REACTOR PROJECTS SECTION, U.S. NUCLEAR REGULATORY COMMISSION, REGION I

My responsibility in this position was to plan and coordinate the NRC inspection program at three nuclear power plants. This included assessing the results of inspections performed in all areas related to plant construction and operation, e.g. operations, maintenance, surveillance and testing, health physics, emergency planning, and allegations. I was responsible for monitoring day-to-day activities at two operating sites and the completion of construction and pre-operational testing at a third site. I supervised eight engineering inspectors, including six stationed at the nuclear plant sites, and I coordinated activities with other Regional Divisions and Headquarters. The position involved performing frequent systems and plant transient evaluations and assessing licensee performance (SALP).

SEPT 1980 to SEPT 1984 — STRUCTURAL ENGINEER, U.S. NUCLEAR REGULATORY COMMISSION, OFFICE OF NUCLEAR REGULATORY RESEARCH

In this position I was responsible for planning, implementing and monitoring research programs related to materials, fracture and structural integrity of reactor components. I managed over $4 million per year in research contracts. Programs I managed included development of experimental data and analytic techniques for fracture and leak-before-break analyses of piping systems, development of data for assessing the susceptibility of piping to stress corrosion cracking and development of probabilistic fracture mechanics methods for pressure vessels and piping. I wrote the VISA (Vessel Integrity Simulation Analysis) computer code and performed extensive mechanistic and probabilistic analyses of reactor vessels subject to pressure and thermal transients. While in this position I helped to develop the regulatory screening criteria for pressurized thermal shock, and I was a member of the pipe crack study group that recommended the regulatory positions on pipe cracking and postulated pipe breaks.

APRIL 1976 to SEPT 1980	APPLIED MECHANICS ENGINEER AND TASK MANAGER U.S. NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION

My primary responsibilities in this position were reviewing and evaluating safety issues related to mechanical, materials and structural aspects of operating commercial reactors and preparing safety evaluation reports to support licensing activities. I performed engineering reviews of issues related to steam generators, piping, pressure vessels, Mark I containments, spent fuel pool expansions and other plant modifications. During the period 1978 to 1980, I was Task Manager for the NRC Generic Safety Issue on steam generator tube integrity. I was responsible for planning, organizing and implementing the Task Action Plan to resolve steam generator tube degradation safety issues. During this period there were eight engineers assigned to the program. I was responsible for coordinating, monitoring and evaluating their work. I was also a member of the task group for the resolution of the unresolved safety issue on reactor vessel low upper shelf fracture toughness.

EDUCATION:

Bachelors Degree in in Engineering Mechanics	University of Missouri at Rolla	1974
Masters Degree in in Engineering Mechanics	University of Missouri at Rolla	1976
Graduate Certificate in Technology & Administration	American University Washington, D.C.	1979
Masters Degree in Business Administration	University of Maryland College Park, Maryland	1982

Required NRC Supervisory Training and Numerous NRC Technical Training Courses.

PROFESSIONAL QUALIFICATIONS

Christopher Ivan Grimes
Chief, Technical Specifications Branch
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

EDUCATION

Bachelor of Science degree in Nuclear Engineering, Oregon State University, 1971. Post-graduate studies in Nuclear Engineering at Catholic University, Washington, D.C.

EXPERIENCE

April 1991 to Present, Chief of the Technical Specifications Branch with the Nuclear Regulatory Commission. Responsible for supervising the development and maintainance of standard technical specifications, based on new regulatory requirements, new technical considerations, operating experience, and the Commission's policy statement on technical specification improvements. Also responsible for establishing policies and programs for the development of implementation guidance for technical specifications, development of interpretations of technical specification requirements, development of technical specifications for license applications or major license upgrades, and assistance in the screening of license amendment applications for generic applicability.

June 1990 to April 1991, Director of Project Directorate IV-2 with the Nuclear Regulatory Commission. Responsible for directing the licensing activities related to four of the nuclear power plants located in the NRC's Region IV. These responsibilities consisted of managing the overall safety and environmental assessment for the assigned plants, monitoring daily operations, coordinating technical reviews and licensing actions, and assisting Region IV in the routine and special inspection activities.

March 1987 to June 1990, Director of the Comanche Peak Project Division with the Nuclear Regulatory Commission. Responsible for directing all of the licensing activities, including the technical review and inspection efforts, related to the operating license application for the Comanche Peak Steam Electric Station. Also responsible for directing allegation follow-up, coordinating the staff's technical review with the General Counsel's legal activities and the Atomic Safety and Licensing Board's hearing activities, and implementation of the recommendations approved by the Commission in NUREG-1257 concerning the investigation of Region IV management of Comanche Peak inspection activities.

November 1985 to March 1987, Director of the Integrated Safety Assessment Program with the Nuclear Regulatory Commission. Responsible for policy development, supervision, and implementation of the follow-on to the Systematic Evaluation Program of the safety of operating power reactors. Directed a pilot project to manage the licensing activities for two operating nuclear power plants using probabilistic analyses to rank the importance of plant modifications and implementation schedules.

August 1984 to November 1985, Chief of the Systematic Evaluation Program Branch with the Nuclear Regulatory Commission. Responsible for policy development, supervision, and coordination of the NRC's efforts associated with the safety review for the systematic evaluation of operating nuclear power reactors.

April 1982 to August 1984, Section Leader, Systematic Evaluation Program Branch, with the Nuclear Regulatory Commission. Responsible for the direct supervision of integrated assessments conducted under the Systematic Evaluation Program review of several older nuclear power plant designs, compared to current regulatory requirements. Responsible for the development and coordination of related staff technical positions and backfitting recommendations. Appointed as Acting Branch Chief in September 1983.

September 1980 to April 1982, Senior Project Manager with the Nuclear Regulatory Commission. Responsible for directing the staff's technical and environmental reviews of the license applications for the Clinton Nuclear Power Station and the Combustion Engineering Standard Safety Analysis Report (CESSAR) Final Design Application. Also responsible for updating the Commission's standardization policies and developing new licensing procedures; for example, the development and implementation of the Standard Review Plan rule, 10 CFR 50.34(g).

March 1975 to September 1980, Engineering Systems Analyst with the Nuclear Regulatory Commission. Responsible for the evaluation of nuclear power plant operating experience and proposed changes in plant design and operation, associated with plant support systems. Assigned as task leader for the evaluation and analysis of suppression pool hydrodynamic loads which were not considered in the original design of the General Electric Mark I containment; responsible for the coordination of related technical reviews and research activities.

November 1973 to March 1975, Reactor Engineer with the Atomic Energy Commission. Responsible for the conduct of technical reviews of the containment systems for proposed nuclear power plant designs, proposed changes to the design and operation of licensed nuclear power plants, and the evaluation of plant operating experience.

February 1972 to November 1973, Manufacturing Engineer with Nuclear Engineering and Components, Inc. of Santa Clara, California, and Creative Industries of Campbell, California. Responsible for the manufacturing, testing, and quality control certification of valves and monitoring instruments for nuclear power, computer, and aerospace applications.

ATTACHMENT 1

Standard Practice for

CONDUCTING SURVEILLANCE TESTS FOR LIGHT-WATER COOLED NUCLEAR POWER REACTOR VESSELS, E 706 (IF)[1]

This standard is issued under the fixed designation E 185; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

ε1 Note—Section 9.2.3 was corrected editorially and the designation date was changed July 1, 1982.
ε2 Note—The title was changed editorially in July 1985.

1. Scope

1.1 This practice covers procedures for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water cooled nuclear power reactor vessels. This practice includes guidelines for designing a minimum surveillance program, selecting materials, and evaluating test results.

1.2 This practice was developed for all light-water cooled nuclear power reactor vessels for which the predicted maximum neutron fluence (E > 1 MeV) at the end of the design lifetime exceeds 1×10^{21} n/m² (1×10^{17} n/cm²) at the inside surface of the reactor vessel.

2. Applicable Documents

2.1 *ASTM Standards:*

A 370 Methods and Definitions for Mechanical Testing of Steel Products[2]

E 8 Methods of Tension Testing of Metallic Materials[3]

E 21 Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials[3]

E 23 Methods for Notched Bar Impact Testing of Metallic Materials[3]

E 208 Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels[3]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance[4]

E 560 Recommended Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results[4]

2.2 *American Society of Mechanical Engineers Standard:* Boiler and Pressure Vessel Code, Sections III and XI[5]

3. Significance and Use

3.1 Predictions of neutron radiation effects on pressure vessel steels are considered in the design of light-water cooled nuclear power reactors. Changes in system operating parameters are made throughout the service life of the reactor vessel to account for radiation effects. Because of the variability in the behavior of reactor vessel steels, a surveillance program is warranted to monitor changes in the properties of actual vessel materials caused by long-term exposure to the neutron radiation and temperature environment of the given reactor vessel. This practice describes the criteria that should be considered in planning and implementing surveillance test programs and points out precautions that should be taken to ensure that: (*1*) capsule exposures can be related to beltline exposures, (*2*) materials selected for the surveillance program are samples of those materials most likely to limit the operation of the reactor vessel, and (*3*) the tests yield results useful for the evaluation of radiation effects on the reactor vessel.

[1] This practice is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications.

Current edition approved July 1, 1982. Published September 1982. Originally published as E 185 – 61 T. Last previous edition E 185 – 79.

[2] *Annual Book of ASTM Standards*, Vol 01.04.

[3] *Annual Book of ASTM Standards*, Vol 03.01.

[4] *Annual Book of ASTM Standards*, Vol 12.02.

[5] Available from the American Society of Automotive Engineers, 345 E. 47th St., New York, NY 10017.

3.2 The design of a surveillance program for a given reactor vessel must consider the existing body of data on similar materials in addition to the specific materials used for that reactor vessel. The amount of such data and the similarity of exposure conditions and material characteristics will determine their applicability for predicting the radiation effects. As a large amount of pertinent data becomes available it may be possible to reduce the surveillance effort for selected reactors by integrating their surveillance programs.

4. Definitions

4.1 *adjusted reference temperature*—the reference temperature adjusted for irradiation effects by adding to RT_{NDT} the transition temperature shift (see 4.15).

4.2 *base metal (parent material)*—as-fabricated plate material or forging material other than a weldment or its corresponding heat-affected-zone (HAZ).

4.3 *beltline*—the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core, and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material.

4.4 *EOL*—end-of-life; the design lifetime in terms of years; effective full power years, or neutron fluence.

4.5 *index temperature*—that temperature corresponding to a predetermined level of absorbed energy, lateral expansion, or fracture appearance obtained from the average (best fit) Charpy transition curve.

4.6 *fraction strength*—in a tensile test, the load at fracture divided by the initial cross-sectional area of the test specimen.

4.7 *fracture stress*—in a tensile test, the load at fracture divided by the cross-sectional area of the test specimen at time of fracture.

4.8 *heat-affected zone* (HAZ)—plate material or forging material extending outward from, but not including, the weld fusion zone in which the microstructure of the base metal has been altered by the heat of the welding process.

4.9 *lead factor*—the ratio of the neutron flux density at the location of the specimens in a surveillance capsule to the neutron flux density at the reactor pressure vessel inside surface at the peak fluence location.

4.10 *neutron fluence*—the time integrated neutron flux density, expressed in neutrons per square metre or neutrons per square centimetre.

4.11 *neutron flux density*—a measure of the intensity of neutron radiation within a given range of neutron energies; the product of the neutron density and velocity, measured in neutrons per square metre-second or neutrons per square centimetre-second.

4.12 *neutron spectrum*—the distribution of neutrons by energy levels impinging on a surface, which can be calculated based on analysis of multiple neutron dosimeter measurements, on the assumption of a fission spectrum, or from a calculation of the neutron energy distribution.

4.13 *nil-ductility transition temperature* (T_{NDT})—the maximum temperature at which a standard drop weight specimen breaks when tested in accordance with Method E 208.

4.14 *reference temperature* (RT_{NDT})—See subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

4.15 *transition temperature shift* (ΔRT_{NDT}) *or adjustment of reference temperature*—the difference in the 41-J (30-ft·lbf) index temperatures from the average Charpy curves measured before and after irradiation.

4.16 *transition region*—the region on the transition temperature curve in which toughness increases rapidly with rising temperature. In terms of fracture appearance, it is characterized by a rapid change from a primarily cleavage (crystalline) fracture mode to primarily shear (fibrous) fracture mode.

4.17 *Charpy transition curve*—a graphic presentation of Charpy data, including absorbed energy, lateral expansion, and fracture appearance, extending over a range including the lower shelf energy (< 5 % shear), transition region, and the upper shelf energy (> 95 % shear).

4.18 *upper shelf energy level*—the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy.

5. Test Materials

5.1 *Materials Selection*:

5.1.1 Surveillance test materials shall be prepared from samples taken from the actual materials used in fabricating the beltline of the reactor vessel. These surveillance test materials shall include one heat of the base metal, one butt weld, and one weld heat-affected-zone (HAZ). The base metal, weld metal, and HAZ (Note 1) materials included in the program shall be those predicted to be most limiting, with regard to setting pressure-temperature limits, for operation of the reactor to compensate for radiation effects during its lifetime (Note 2). The beltline materials shall be evaluated on the basis of initial reference temperature (RT_{NDT}), the predicted changes in the initial properties as a function of chemical composition (for example, copper (Cu) and phosphorus (P)) (Note 3), and the neutron fluence during reactor operation.

NOTE 1—The base metal for the weld heat-affected-zone (HAZ) to be monitored shall correspond to one of the base metals selected for the surveillance program.

NOTE 2—The data used for the selection of surveillance test materials shall be that obtained in accordance with ASME Code Section III requirements.

NOTE 3—Other residual/alloy elements such as Ni, Si, Mn, Mo, Cr, C, S, and V may contribute to overall radiation behavior of ferritic materials.

5.1.2 The base metal and the weld with the highest adjusted reference temperature at end-of-life shall be selected for the surveillance program. If the Charpy upper shelf energy of any of the beltline materials is predicted to drop to a marginal level (currently considered to be 68 J (50 ft·lbf) at the quarter thickness (¼ T) location) during the operating lifetime of the vessel, provisions shall be made to also include that material in the surveillance program, preferably in the form of fracture toughness specimens. These additional specimens may be substituted in part for specimens of the material least likely to be limiting.

5.1.3 The adjusted reference temperature of the materials in the reactor vessel beltline shall be determined by adding the appropriate values of transition temperature shift to the reference temperature of the unirradiated material. The transition temperature shift and Charpy upper shelf energy drop can be determined from relationships of fluence and chemical composition.

5.4 *Material Sampling*—A minimum test program shall consist of the material selected in 5.1, taken from the following locations: (*1*) base metal from one plate or forging used in the beltline, (*2*) weld metal made with the same heat of weld wire and lot of flux and by the same welding practice as that used for the selected beltline weld, and (*3*) the heat-affected-zone associated with the base metal noted above.

5.5 *Archive Materials*—Representative test stock to fill at least two additional capsules with test specimens of the base metal, weld, and heat-affected-zone materials used in the program shall be retained with full documentation and identification. It is recommended that this test stock be in the form of full-thickness sections of the original materials (plates, forgings, and welds).

5.6 *Fabrication History*—The fabrication history (austenitizing, quench and tempering, and post-weld heat treatment) of the test materials shall be fully representative of the fabrication history of the materials in the beltline of the reactor vessel and shall be recorded.

5.7 *Chemical Analysis Requirements*—The chemical analysis required by the appropriate product specifications for the surveillance test materials (base metal and as-deposited weld metal) shall be recorded and shall include phosphorus (P), sulfur (S), copper (Cu), vanadium (V), and nickel (Ni), as well as all other alloying and residual elements commonly analyzed for in low-alloy steel products. The product analysis shall be verified by analyzing a minimum of three test specimens randomly selected from both the base metal and the as-deposited weld metal.

6. Test Specimens

6.1 *Type of Specimens*—Charpy V-notch impact specimens corresponding to the Type A specimen described in Methods A 370 and E 23 shall be used. The gage section of irradiated and unirradiated tension specimens shall be of the same size and shape. Tension specimens of the type, size, and shape described in Methods A 370 and E 8 are recommended. Additional fracture toughness test specimens shall be employed to supplement the information from the Charpy V-notch specimens if the surveillance materials are predicted to exhibit marginal properties.

6.2 *Specimen Orientation and Location*—Tension and Charpy specimens representing the base metal and the weld heat-affected-zone shall be removed from about the quarter-thickness (¼ T) locations. Material from the [redacted] for test specimens. Specimens representing weld metal may be removed at all locations throughout the thickness with the exception of locations within 12.7 mm (½ in.) of the root or surfaces of the welds. The tension and Charpy specimens from base metal shall be oriented so that the major axis of the specimen is parallel to the surface and normal to the principal rolling direction for plates, or normal to the major working direction for forgings as described in Section III of the ASME Code. The axis of the notch of the Charpy specimen for base metal and weld metal shall be oriented perpendicular to the surface of the material; for the HAZ specimens, the axis of the notch shall be as close to perpendicular to the surface as possible so long as the entire length of the notch is located within the HAZ. The recommended orientation of the weld metal and HAZ specimens is shown in Fig. 1. Weld metal tension specimens may be oriented in the same direction as the Charpy specimens provided that the gage length consists entirely of weld metal. The weldment shall be etched to define the weld heat-affected-zone. The notch roots in the HAZ Charpy specimens shall be at a standard distance of approximately 0.8 mm (1/32 in.) from the weld fusion line. The orientation of the HAZ samples with respect to the major working direction of the parent material shall be recorded.

6.3 *Quantities of specimens*:

6.3.1 *Unirradiated Baseline Specimens*—It is recommended that 18 Charpy specimens be provided, of which a minimum of 15 specimens shall be tested to establish a full transition temperature curve for each material (base metal, HAZ, weld metal). The three remaining Charpy specimens should be reserved to provide supplemental data in instances such as excessive data scatter. At least three tension test specimens shall be provided to establish the unirradiated tensile properties for base metal and weld metal.

6.3.2 *Irradiated Specimens*—The minimum number of test specimens for each irradiation exposure set (capsule) shall be as follows:

Material	Charpy	Tension
Base metal	12	3
Weld metal	12	3
HAZ	12	—

It is suggested that a greater quantity of the above specimens be included in the irradiation capsules whenever possible.

7. Irradiation Requirements

7.1 *Encapsulation of Specimens*—Specimens should be maintained in an inert environment within a corrosion-resistant capsule to prevent deterioration of the surface of the specimens during radiation exposure. Care should be exercised in the design of the capsule to ensure that the temperature history of the specimens duplicates, as closely as possible, the temperature experienced by the reactor vessel. Surveillance capsules should be sufficiently rigid to prevent mechanical damage to the specimens and monitors during irradiation. The design of the capsule and capsule attachments shall also permit insertion of replacement capsules into the reactor vessel if required at a later time in the lifetime of the vessel. The design of the capsule holder and the means of attachment shall (*1*) preclude structural material degradation by the attachment welds, (*2*) avoid interference with inservice inspection required by ASME Code Section XI, and (*3*) ensure the integrity of the capsule holder during the service life of the reactor vessel.

7.2 *Location of Capsules*:

7.2.1 *Vessel Wall Capsules (Required)*—Surveillance capsules shall be located within the reactor vessel so that the specimen irradiation history duplicates as closely as possible, within the physical constraints of the system, neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel. It is recommended that the surveillance capsule lead factors (the ratio of the instantaneous neutron flux density at the specimen location to the maximum calculated neutron flux density at the inside surface of the reactor vessel wall) be in the range of one to three. This range of lead factors will minimize the calculational uncertainties in extrapolating the surveillance measurements from the specimens to the reactor vessel wall and maximize the ability of the program to monitor material property changes throughout the life of the reactor vessel.

7.2.2 *Accelerated Irradiation Capsules (Optional)*—Additional test specimens may be positioned at locations closer to the core than those described in 7.2.1 for accelerated irradiation.

7.3 *Neutron Dosimeters:*

7.3.1 *Selection of Neutron Dosimeters*—Neutron dosimeters for the surveillance capsules shall be selected according to Guide E 482. The group of monitors selected shall be capable of providing fast neutron fluence, fast neutron spectrum, and thermal neutron flux density information. Dosimeters shall be included in every capsule.

7.3.2 *Location of Neutron Dosimeters*—Dosimeters shall be located within the vessel wall capsules (7.2.1) and the accelerated capsules (7.2.2) if used.

7.3.3 Separate dosimeter capsules should also be used to monitor radiation conditions independent of the specimen capsules if it is expected that the withdrawal schedule will otherwise result in saturation of the dosimeter activities.

7.4 *Correlation Monitors (Optional):*

7.4.1 *Selection of Correlation Monitor Materials*—Correlation monitors[6] have been found to be useful as an independent check on the measurement of irradiation conditions for the surveillance materials. Correlation monitor materials should be well characterized in terms of irradiation behavior (transition temperature shift). The magnitude of the transition temperature shift for this material should be measureable for the selected exposures.

7.5 *Temperature Monitors:*

7.5.1 *Selection of Temperature Monitors*—Major differences between specimen irradiation temperature and design temperature, occuring as a result of capsule design features, variation in reactor coolant temperature, or both, can affect the extent of radiation induced property changes in the surveillance materials. Since it is not practical to instrument the surveillance capsules, low melting point elements or eutectic alloys are used instead as monitors to detect significant variations in exposure temperature. These monitors are used in surveillance programs to provide evidence of the maximum exposure temperature of the specimens. The monitor materials should be selected to indicate unforeseen capsule temperatures.

7.5.2 *Location of Temperature Monitors*—One set of temperature monitors shall be located within the capsule where the specimen temperature is predicted to be the maximum. Additional sets of temperature monitors may be placed at other locations within the capsule to characterize the temperature profile.

7.6 *Number of Surveillance Capsules and Withdrawal Schedule.*

7.6.1 *Number of Capsules*—A sufficient number of surveillance capsules shall be provided to monitor the effects of neutron irradiation on the reactor vessel throughout its operating lifetime. The basis for the number of capsules to be installed at beginning of life is the predicted transition temperature shift, as shown in Table 1. The decrease in the upper shelf energy may also be a factor (see 5.1, 5.2, and 5.3). Additional capsules may be needed to monitor the effect of a major core change or annealing of the vessel, or to provide supplemental toughness data for evaluating a flaw in the beltline. It is recommended that full-thickness sections of material be kept instead of loaded capsules, because the preferred type and size of test specimen may change in the intervening years. The archive material required in 5.5 is to be used for the additional capsules.

7.6.2 *Withdrawal Schedule*—The capsule withdrawal schedule should permit monitoring of long-time effects which are difficult to achieve in test reactors. Table 1 lists the recommended number of capsules and the withdrawal schedule for three ranges of predicted transition temperature shift. The withdrawal schedule is in terms of effective full-power years (EFPY) of the vessel with a design life of 32 EFPY. Other factors that must be considered in establishing the withdrawal schedule are presented in Table 1. The first capsule is scheduled for withdrawal early in the vessel life to verify the initial predictions of the surveillance material response to the actual radiation environment. It is removed when the predicted shift exceeds the expected scatter by sufficient margin to be measureable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal will permit verification of the adequacy and conservatism of the reactor vessel pressure/temperature operational limits. The withdrawal schedule of the final two capsules is adjusted by the lead factor so the exposure of the second to last capsule does not exceed the peak end-of-life (EOL) fluence on the inside surface of the vessel, and so the exposure of the final capsule does not exceed twice the EOL vessel inside surface peak fluence. The decision on when to test specimens from the final capsule need not be made until the results from the preceding capsules are known.

7.6.3 *Implementation of Table 1:*

7.6.3.1 Estimate the peak vessel inside surface fluence at EOL and the corresponding transition temperature shift. This identifies the number of capsules required.

7.6.3.2 Estimate the lead factor for each surveillance capsule relative to the peak beltline fluence.

7.6.3.3 Calculate the number of EFPY for the capsule to reach the peak vessel EOL fluence at the inside surface and ¼ T locations. These are used to establish the withdrawal schedule for all but the first capsule.

7.6.3.4 Schedule the capsule withdrawals at the nearest vessel refueling date.

8. Measurement of Radiation Exposure Conditions

8.1 *Temperature Environment*—The maximum exposure temperature of the surveillance capsule materials shall be determined. If a discrepancy (> 14°C or 25°F) occurs between the observed and the expected capsule exposure temperatures, an analysis of the operating conditions shall be conducted to determine the magnitude and duration of these differences.

8.2 *Neutron Irradiation Environment:*

8.2.1 The neutron flux density, neutron energy spectrum, and neutron fluence of the surveillance specimens and the corresponding maximum values for the reactor vessel shall be determined in accordance with the guidelines in Guide E 482 and Recommended Practice E 560.

8.2.2 The specific method of determination shall be documented.

8.2.3 Neutron flux density and fluence values (E > 0.1 and 1 MeV) shall be determined and recorded using both a calculated spectrum and an assumed fission spectrum.

9. Measurement of Mechanical Properties

9.1 *Tension Tests:*

9.1.1 *Method*—Tension testing shall be conducted in accordance with Methods E 8 and Recommended Practice E 21.

9.1.2 *Test Temperature:*

9.1.2.1 *Unirradiated*—The test temperatures for each material shall include room temperature, service temperature, and one intermediate temperature to define the strength *versus* temperature relationship.

9.1.2.2 *Irradiated*—One specimen from each material shall be tested at a temperature in the vicinity of the upper end of the Charpy energy transition region. The remaining specimens from each material shall be tested at the service temperature and the midtransition temperature.

9.1.3 *Measurements*—For both unirradiated and irradiated materials, determine yield strength, tensile strength, fracture load, fracture strength, fracture stress, total and uniform elongation, and reduction of area.

9.2 *Charpy Tests:*

9.2.1 *Method*—Charpy tests shall be conducted in accordance with Methods E 23 and A 370.

9.2.2 *Test Temperature:*

9.2.2.1 *Unirradiated*—Test temperatures for each material shall be selected to establish a full transition temperature curve. One specimen per test temperature may be used to define the overall shape of the curve. Additional tests should be performed in the region where the measurements described in 9.2.3 are made.

9.2.2.2 *Irradiated*—Specimens for each material will be tested at temperatures selected to define the full energy transition curve. Particular emphasis should be placed on defining the 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperatures and the upper shelf energy.

9.2.3 *Measurements*—For each test specimen, measure the impact energy, lateral expansion, and percent shear fracture appearance. From the unirradiated and irradiated transition temperature curves determine the 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperatures and the upper shelf energy. The index temperatures

[6] Information regarding the availability of correlation monitors can be obtained from ASTM Committee E-10. See also ASTM DS54, July 1974.

and the upper shelf energy shall be determined from the average curves.

9.2.3.1 Obtain from the material qualification test report the initial reference temperature (RT_{NDT}) as defined in the ASME Code, Section III, Subarticle NB 2300 for unirradiated materials.

9.3 *Hardness Tests (Optional)*—Hardness tests may be performed on unirradiated and irradiated Charpy specimens. The measurements shall be taken in areas away from the fracture zone or the edges of the specimens. The tests shall be conducted in accordance with Methods A 370.

9.4 *Supplemental Tests (Optional)*—If supplemental fracture toughness tests are conducted (in addition to tests conducted on tension and Charpy specimens as described in 6.1) the test procedures shall be documented.

9.5 *Calibration of Equipment*—Procedures shall be employed assuring that tools, gages, recording instruments, and other measuring and testing devices are calibrated and properly adjusted periodically to maintain accuracy within necessary limits.[7] Whenever possible calibration shall be conducted with standards traceable to the National Bureau of Standards. Calibration status shall be maintained in records traceable to the equipment.

10. Determination of Irradiation Effects

10.1 *Tension Test Data:*

10.1.1 Determine the amount of radiation strengthening by comparing unirradiated test results with irradiated test results at the temperatures specified in 9.1.2.

10.1.2 The tensile strength data can be verified using the results from the hardness test (optional) described in 9.3.

10.2 *Charpy Test Data:*

10.2.1 Determine the radiation induced transition temperature shifts by measuring the difference in the 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperatures before and after irradiation. The index temperatures shall be obtained from the average curves.

10.2.2 Determine the adjusted reference temperature by adding the shift corresponding to the 41-J (30-ft·lbf) index determined in 10.2.1 to the initial reference temperature obtained in 9.2.3.1.

10.2.3 Determine the radiation induced change in the upper shelf energy (USE) from measurements made before and after irradiation using average value curves.

10.2.4 *(Optional)*—Determine the radiation induced change in temperature corresponding to 50 % of the upper shelf energy before and after irradiation from average value curves.

10.3 *Supplemental Test Data (Optional)*—If additional, supplemental tests are performed (9.4), the data shall be recorded to supplement the information from the tensile and Charpy tests.

10.4 *Retention of Test Specimens*—It is recommended that all broken test specimens be retained until released by the owner in the event that additional analyses are required to explain anomalous results.

11. Report

11.1 The following information shall be provided. This report shall consist of the following elements. Where applicable, both SI units and conventional units shall be reported.

11.2 *Surveillance Program Description*—Description of the reactor vessel including the following:

11.2.1 Location of the surveillance capsules with respect to the reactor vessel, reactor vessel internals, and the reactor core.

11.2.2 Location in the vessel of the plates or forgings and the welds.

11.2.3 Location(s) of the peak vessel fluence.

11.2.4 Lead factors between the specimen fluence and the peak vessel fluence at the I.D. and the ¼ T locations.

11.2.5 *Surveillance Material Selection:*

11.2.5.1 Description of all beltline materials including chemical analysis, fabrication history, Charpy data, tensile data, drop-weight data and initial RT_{NDT}.

11.2.5.2 Describe the basis for selection of surveillance materials.

11.3 *Surveillance Material Characterization:*

11.3.1 Description of the surveillance material including fabrication history, material source (heat or lot), and any differences between the surveillance material history and that of the reactor vessel material history.

[7] Standardized specimens for certification of Charpy impact machines are available from the Army Materials and Mechanics Research Center, Watertown, MA 02172, Attn: DRXMR-MQ.

11.3.2 Location and orientation of the test specimens in the parent material.

11.3.3 *Test Specimen Design:*

11.3.3.1 Description of the test specimens (tension, Charpy, and any other types of specimens used), neutron dosimeters, and temperature monitors.

11.3.3.2 Certification of calibration of all equipment and instruments used in conducting the tests.

11.4 *Test Results:*

11.4.1 *Tension Tests:*

11.4.1.1 Trade name and model of the testing machine, gripping devices, extensometer, and recording devices used in the test.

11.4.1.2 Speed of testing and method of measuring the controlling testing speed.

11.4.1.3 Complete stress-strain curve (if a group of specimens exhibits similar stress-strain curves, a typical curve may be reported for the group).

11.4.1.4 Test data from each specimen as follows:

(*1*) Test temperature;

(*2*) Yield strength or yield point and method of measurement;

(*3*) Tensile strength;

(*4*) Fracture load, fracture strength, and fracture stress;

(*5*) Uniform elongation and method of measurement;

(*6*) Total elongation;

(*7*) Reduction of area; and

(*8*) Specimen identification.

11.4.2 *Charpy Tests:*

11.4.2.1 Trade name and model of the testing machine, available hammer energy capacity and striking velocity, temperature conditioning and measuring devices, and a description of the procedure used in the inspection and calibration of the testing machine.

11.4.2.2 Test data from each specimen as follows:

(*1*) Temperature of test;

(*2*) Energy absorbed by the specimen in breaking, reported in joules (and foot-pound-force);

(*3*) Fracture appearance;

(*4*) Lateral expansion; and

(*5*) Specimen identification.

11.4.2.3 Test data for each material as follows:

(*1*) Charpy 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperature of unirradiated material and of each set of irradiated specimens, along with the corresponding temperature increases for these specimens;

(*2*) Upper shelf energy (USE) absorbed before and after irradiation;

(*3*) Initial reference temperature; and

(*4*) Adjusted reference temperature.

11.4.3 *Hardness Tests (Optional):*

11.4.3.1 Trade name and model of the testing machine.

11.4.3.2 Hardness data.

11.4.4 *Other Fracture Toughness Tests:*

11.4.4.1 If additional tests are performed, the test data shall be reported together with the procedures used for conducting the tests and analysis of the data.

11.4.5 *Temperature and Neutron Radiation Environment Measurements:*

11.4.5.1 Temperature monitor results and an estimate of maximum capsule exposure temperature.

11.4.5.2 Neutron dosimeter measurements, analysis techniques, and calculated results including the following:

(*1*) Neutron flux density, neutron energy spectrum, and neutron fluence in terms of neutrons per square metre and neutrons per square centimetre (> 0.1 and 1 MeV) for the surveillance specimens using both calculated spectrum and assumed fission spectrum assumptions.

(*2*) Description of the methods used to verify the procedures including calibrations, cross sections, and other pertinent nuclear data.

11.5 *Application of Test Results:*

11.5.1 Extrapolation of the neutron flux and fluence results to the surface and ¼ T locations of the reactor vessel at the peak fluence location.

11.5.2 Comparison of fluence determined from the dosimetry analysis with original predicted values.

11.5.3 Extrapolation of fracture toughness properties to the surface and ¼ T locations of the reactor vessel at the peak fluence location.

11.6 *Deviations*—Deviations or anomalies in procedure from this practice shall be identified and described fully in the report.

TABLE 1 Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule (Schedule in Terms of Effective Full-Power Years of the Reactor Vessel)

	Predicted Transition Temperature Shift at Vessel Inside Surface		
	≤ 56°C (≤ 100°F)	> 56°C (> 100°F) ≤ 111°C (≤ 200°F)	> 111°C (> 200°F)
Minimum Number of Capsules	3	4	5
Withdrawal Sequence:			
First	6[A]	3[A]	1.5[A]
Second	15[B]	6[C]	3[D]
Third	EOL[E]	15[B]	6[C]
Fourth		EOL[E]	15[B]
Fifth			EOL[E]

[A] Or at the time when the accumulated neutron fluence of the capsule exceeds 5×10^{22} n/m² (5×10^{18} n/cm²), or at the time when the highest predicted ΔRT_{NDT} of all encapsulated materials is approximately 28°C (50°F), whichever comes first.

[B] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

[C] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel ¼ T location, whichever comes first.

[D] Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

[E] Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.

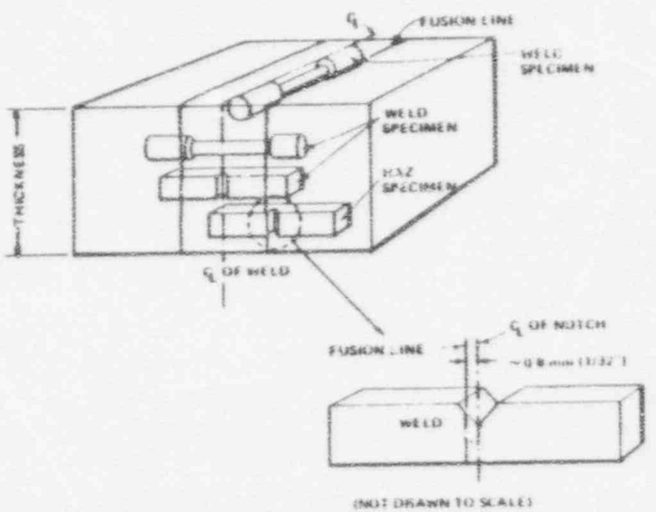

FIG. 1 Location of Test Specimens Within Weld and Heat-Affected-Zone (HAZ) Test Material

ATTACHMENT 2

February 25, 1983

SECY-83-80

RULEMAKING ISSUE
(Affirmation)

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: 10 CFR PART 50--GENERAL REVISION OF APPENDICES G AND H, FRACTURE TOUGHNESS AND REACTOR VESSEL MATERIAL SURVEILLANCE REQUIREMENTS

Purpose: Obtain Commission approval of a notice of final rulemaking.

Issue: Modification of NRC regulations involving the requirements for fracture toughness of the reactor coolant pressure boundary, including surveillance of neutron radiation embrittlement of the reactor vessel beltline materials.

Discussion: Appendix G, "Fracture Toughness Requirements", and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," have undergone only limited revision in over nine years of use. In this general revision, the requirements of Appendices G and H have been updated to be more consistent with current technology and pertinent National Standards. Some of the amendments are intended to clarify the applicability of these requirements to older plants; that is, those built to ASME Codes earlier than the Summer 1972 Addenda to the 1971 Edition, which often requires consideration of proposed alternatives to specific requirements. The amendments specify when acceptance of a proposed alternative must take the form of an exemption granted by the Commission and when acceptance may be granted by the Director of Nuclear Reactor Regulation as being equivalent to the NRC requirements. Two of the amendments modify requirements that have proved to be unduly conservative. A number of other amendments shorten and simplify these regulations by replacing technical detail with references to the ASME Boiler and Pressure Vessel Code and to ASTM E 185, "Standard Practice for Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels."

The notice of proposed rulemaking on this issue was published for public comment in the FEDERAL REGISTER on

Contact: P. N. Randall
443-5903

November 14, 1980. Thirteen replies were received from utilities and vendors concerned with the application of specific requirements. An analysis of the comments received and the staff response is given in Enclosure 4, and a summary is given in the Supplementary Information section of Enclosure 1.

Recommendation: That the Commission:

1. Approve publication of the amendments to Appendices G and H to 10 CFR Part 50 (Enclosure 1) as a final rule.

2. Note the staff conclusions set forth in Enclosure 3, which provides the analysis called for by the Periodic and Systematic Review established by Task IV.G.2. of the TMI Action Plan.

3. Certify that this rule will not have a significant economic impact on a substantial number of small entities, in order to satisfy requirements of the Regulatory Flexibility Act, 5 U.S.C. 605(b).

4. Note:

   a. That the amendments to 10 CFR Part 50 will be published in the Federal Register, and will become effective 60 days after publication.

   b. No environmental impact statement, negative declaration, or environmental impact appraisal need be prepared in connection with the amendments because the action taken by the amendments will not significantly affect the quality of the human environment.

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

   d. The Office of Public Affairs concurs that a public announcement is not needed.

   e. The NRC staff will inform the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs, the Subcommittee on Energy and Power of the House Committee on Interstate and Foreign Commerce, the Subcommittee on Environment, Energy and Natural Resources of the House Committee on Government Operations, and the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works of this action by letter such as Enclosure 5.

f. The Federal Register notice of rulemaking will be distributed by ADM to power reactor licensees/ permit holders, applicants for a construction permit for a power reactor, public interest groups, and nuclear steam system suppliers.

g. The Chief Counsel for Advocacy of the Small Business Administration will be informed by DRR of the certification regarding economic impact on small entities together with the reason for it.

h. Although this rule does not involve a significant question of policy, action by the Commission is required since the final amendments would modify current policy concerning the granting of exemptions by the Commission by providing that certain alternative methods for meeting the requirements in Appendices G and H to Part 50 may be approved by the Director, Office of Nuclear Reactor Regulation.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Federal Register Notice
2. Regulatory Analysis Statement
3. Analysis with respect to the periodic and systematic review of regulations
4. Analysis of public comments and staff response
5. Draft Congressional Letter

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Monday, March 14, 1983.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Monday, March 7, 1983, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of March 14, 1983. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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ENCLOSURE 1

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:

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 wirtten comment by January 13, 1981. Thirteen replies were received. All of the replies were from utilities or vendors concerned with the application of specific requirements. 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 signifiant technical question, which will require rechecking 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 also during 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 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.

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, D.C. 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, D.C. 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.

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.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for 10 CFR Part 50 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 189, 186, 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). 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 issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)), §§ 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 public 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.

3. In section 50.55a, paragraph (i) is deleted and paragraph (j) is redesignated paragraph (i).

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

§ 50.60 - Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all light-water 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.

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

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

[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.

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.

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 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 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 Appendx.

1. That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirments 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 (insert the effective data of this amendment), 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 (insert the effective date of this amendment), 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 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 jeopeardized by operation at reduced power level or b 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 ____________ this _____ day of ____________ 1983.

For the Nuclear Regulatory Commission.

Samuel J. Chilk
Secretary of the Commission

ENCLOSURE 2

REGULATORY ANALYSIS

REVISION OF APPENDICES G AND H,
FRACTURE TOUGHNESS AND SURVEILLANCE PROGRAM REQUIREMENTS

1. THE PROPOSED ACTION

1.1 Description

Fracture control of the reactor vessel as treated by Appendix G in its present form is accomplished principally by pressure-temperature limits, which provide assurance that the vessel is warm enough to have adequate fracture toughness for the corresponding pressure. Radiation damage is compensated for by increasing the required vessel temperature every few years, based on input from the reactor vessel material surveillance program required by Appendix H. Because there is an upper level beyond which toughness cannot be increased by raising metal temperature, Appendix G also requires a minimum upper-shelf toughness. The revision to Appendix G identifies this requirement more clearly than before.

A major part of the revision of Appendix G is deletion of items now covered in the ASME Code and incorporation of the applicable Code provisions by reference. Similarly, parts of Appendix H are deleted and replaced by references to ASTM E 185. Publication of a new edition, E 185-79, containing much technical detail, has made it possible to shorten Appendix H. Paragraph 50.55a(i), which added Appendices G and H to Part 50, is deleted and a new Section 50.60 is added to take the place of paragraph (i). Language is added to clarify how certain requirements of Appendices G and H apply to "old" plants. New language in § 50.60 and in Appendices G and H distinguishes those proposed alternatives to the described requirements that require an exemption to be granted by the Commission from those alternatives that can be accepted by the Director, Office of Nuclear Reactor Regulation (NRR), as being equivalent to the described requirement. A few requirements that have proven to be unduly conservative are modified. Finally, a number of technical requirements are clarified and updated. The specific revisions are discussed below.

1.2 Need for Proposed Action

The use of Appendices G and H since they were originally promulgated in August 1973, has shown that a number of the requirements need clarification in language. Eight years of use has also shown that certain restrictions such as those described below in paragraph IV.A.3. and paragraph IV.A.4. can and should be modified to improve plant efficiency while still providing an adequate level of safety. Finally, there have been changes in the ASME Code and in ASTM E 185 that need to be reflected in Appendices G and H. The net value of the proposed changes should far outweigh their impact.

SPECIFIC REVISIONS

§ 50.60 The change from § 50.55a to § 50.60 (described above) is of value to the NRC and to users of the regulation simply as an editorial clarification. The change will reduce clutter in § 50.55a and remove present ambiguities in the application of the prefatory language of § 50.55a to Appendices G and H.

Appendix G

¶II.F. Redefinition of "adjusted reference temperature" is of value to both the NRC and to licensees. The change from the 50 ft-lb level to the 30 ft-lb level of Charpy energy at which the transition temperature shift is to be measured as an indicator of radiation damage was made for several reasons. The results of analyses of surveillance data from operating reactors showed that the upper-shelf energy in certain vessels would drop below 50 ft-lb with additional radiation, rendering that criterion invalid. Traditionally, shift has been measured at the 30 ft-lb level. Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 1, April 1977, used the 30 ft-lb limit, because the data base was given in those terms. Fifty ft-lb or 35 mils lateral expansion (whichever gives the greater shift) had been chosen in 1972 when those values became part of the definition of RT_{NDT} in the ASME Code. A recent analysis of the data base has shown that shift measured at the 50 ft-lb level is 5-10 percent larger than the 30 ft-lb value, on the average, but the scatter is such that individual comparisons can go either way. The basis for selecting the criterion should be that it produces an upward adjustment of temperature which will give pressure-temperature limits that maintain the required margin of safety against fracture during heatup and cooldown throughout plant life. Implementation of this criterion requires an extensive correlation of Charpy shift data and fracture toughness data obtained with large specimens, irradiated and unirradiated. There are only a few such correlations at the present time, but they support the use of the traditional 30 ft-lb criterion. The 35 mil lateral expansion criterion was dropped, because it was only used as a double check on the 50 ft-lb criterion.

There will be some impact on the laboratory personnel who are responsible for Charpy test procedures. The change from 50 to 30 ft-lb as the level of Charpy energy at which the transition temperature shift should be measured will cause changes in the choice of test temperatures during Charpy testing of irradiated and unirradiated material. The cost involved in making this change is believed to be negligible.

Although not explicitly stated in the present regulation, the 50 ft-lb measure of shift also acted as a warning that the upper-shelf energy level was becoming marginal. To retain this function in the amended regulation, the 50 ft-lb upper-shelf requirement has been added to Section IV. Its significance as a fracture criterion is still the subject of considerable research, thus the purpose of the requirement is to trigger a fracture analysis that uses supplemental fracture toughness test results.

¶II.G. The new definition of "beltline" is of value to licensees because it reduces unnecessary materials testing. Savings are difficult to estimate--perhaps $2000 for a typical vessel.

§III. Deletion of several detailed requirements for materials testing and recordkeeping and substitution of ASME Code requirements therefor is of value to licensees, because they must follow Code requirements anyway.

Language has been added to remove the need for exemptions to operating licenses with regard to certain materials testing requirements, thus saving considerable staff time. For example, for plants built to an edition of the ASME Code earlier than the Summer 1972 Addenda to the 1971 Edition, the Charpy testing of the reactor vessel materials did not yield an explicit number for the reference temperature, RT_{NDT}, as defined in the Summer 1972 addenda and used in Appendix G of the Code. Valid estimates of RT_{NDT} can be made; however, opinion is divided as to whether the use of estimated values of RT_{NDT} requires exemptions to specific paragraphs in Appendix G. Language has been added to paragraph III.A. to avoid the need for processing exemptions in this case.

¶IV.A.2. There is an impact on any licensees who have not fully considered the possibility that flaws at closure flange regions that are highly stressed by the bolt preload may be governing for the first years of service. Pressure-temperature limits for some plants will need to be revised, but the impact on plant operations is expected to be small. The staff analysis of public comments contains an extensive discussion of the need for this amendment and the alternatives considered in making it.

¶IV.A.3. The amendment to this paragraph lowers the minimum permissible temperature for core criticality at low pressure for boiling water reactors (BWRs). This change is of value to owners of BWRs, because it reduces delays in startup. BWRs cannot use pump heat during startup as effectively as PWRs can, because the elevation head of water in the reactor alone is insufficient to meet the NPSH (net pump suction head) requirements of the pumps at all but the lowest speeds. Hence, pump heat is low until there is steam pressure in the reactor.

The decision to make this change was made following staff review of Topical Report NEDO-21778-A from the General Electric Co. The review concluded[1] that the probability of an overpressure transient that would violate the pressure-temperature limits was very low and would not be increased significantly by making this change. Therefore, although the hypothetical transient might occur at a lower temperature, and thus be a more severe violation of the P-T limits, the staff considered the relaxation of the requirement justified. In their request for this change,[1] the General Electric Co. estimated that it would save as much as $600,000 per year per plant in power replacement costs, because it reduces startup time.

¶IV.A.4. Reducing the minimum temperature required for the initial hydrostatic pressure test is of value to licensees in reducing time and expense of heating the reactor vessel to the test temperature and in improving the working conditions for the inspection personnel who perform the leak test by

[1]See NEDO-21778-A "Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors," December, 1978.

visual inspection. Because there will be no fuel in the reactor, public safety is not affected. Safety of inspection personnel is not reduced to a level below that afforded shop test personnel during the shop hydrostatic test.

¶IV.B. The added emphasis on Charpy upper-shelf energy is of value to the NRC, because it clarifies the fact that there is a safety requirement related to fracture at temperatures where the material is not brittle but may have insufficient resistance to ductile tearing. See discussion under ¶II.F. The fracture analyses required if the 50 ft-lb upper-shelf requirement at end of life cannot be met would affect licensees, but this is not a new requirement.

¶V.B. The additions to this paragraph are of value to the NRC and to applicants, because the additions provide clarification in areas where there have been questions in the past.

¶V.C.1. The proposed amendment clarifies the extent of the area that must receive thorough inspection. It is of value to licensees, because it reduces time and cost and minimizes radiation exposure to inspection personnel for those cases where only a limited part of the beltline (perhaps only a single weld) fails to meet the requirements of ¶V.B.

Appendix H

¶II.B.1. Publication of the 1979 edition of ASTM E 185 made it necessary to amend this paragraph to incorporate by reference ASTM E 185, rather than the 1973 edition of E 185, and to specify the applicability of the various editions of E 185 to different parts of each surveillance program. The 1982 edition corrected a printer's omission in the 1979 edition. This amendment is of value to both the NRC and licensees because there has been considerable expansion of E 185 in the 1979 edition and because deletion of large sections of Appendix H eliminates detailed requirements that are better presented as general criteria and explanatory material in the ASTM Recommended Practice.

¶II.C. The expanded criteria for an integrated surveillance program are of value to licensees, because such a program reduces testing costs and exposure of personnel to radiation. The criteria were developed after staff action to permit an integrated program in a specific situation, which was prompted by the need to reduce radiation exposure to the workmen who would otherwise have been required to make modifications to capsule attachments on vessel internals that were radioactive.

§IV. The reporting requirement is not increased, but a schedule requirement is added that is of value to the NRC and others who need to get the surveillance data in timely fashion.

ENCLOSURE 3

ANALYSIS WITH RESPECT TO THE PERIODIC AND SYSTEMATIC REVIEW OF REGULATIONS
(TMI ACTION PLAN TASK IV.G.2)

SUBJECT: 10 CFR Part 50--General Revision of Appendices G and H, Fracture Toughness and Reactor Vessel Material Surveillance Requirements

Criteria for Periodic and Systematic Review of Regulations	NRC Compliance
1. The amended regulations are needed.	The amended regulation implements NRC's statutory authority under the Energy Reorganization Act, as amended. Sections 202(3) and (4) of the Energy Reorganization Act as amended provide the NRC with licensing and regulatory authority over the construction of nuclear power plants. Appendices G and H provide the basis for the pressure-temperature limits for plants, which are an essential part of their Technical Specifications. The amendments update an existing regulation after 8 years of use to make it more consistent with pertinent National Standards. The amendments will reduce the need for exemptions by specifying when acceptance of a proposed alternative must take the form of an exemption granted by the Commission and when acceptance may be granted by the Director of Nuclear Reactor Regulation. Two of the amendments modify requirements that have proved to be unduly conservative. A number of other amendments shorten and simplify these regulations by replacing technical detail with references to the ASME Boiler and Pressure Vessel Code and to ASTM E 185, "Standard Practice for Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels."
2. The direct and indirect effects of the regulations have been adequately considered	The direct and indirect effects of this rulemaking were considered in the Value/Impact Analysis prepared in connection with the proposed role. (See Enclosure 2).
3. Alternative approaches have been considered and the least burdensome of the acceptable alternatives have been chosen.	One objective of the amendments was to reduce the burden on licensees and staff that is imposed by the present regulation, without reducing margins of safety.

SUBJECT: 10 CFR Part 50--General Revision of Appendices G and H, Fracture Toughness and Reactor Vessel Material Surveillance Requirements

Criteria for Periodic and Systematic Review of Regulations	NRC Compliance
4. Public comments have been considered and an adequate response has been prepared.	See Enclosure 4, "Abstract of Comments and Staff Response to proposed revision to 10 CFR Part 50, Appendices G and H, Fracture Toughness and Reactor Vessel Material Surveillance Program Requirements, Published for Public Comment in the Federal Register November 13, 1980."
5. The regulation is written so that it is understandable to those who must comply with it.	The proposed amendment has been reviewed and edited for the specific purpose of ensuring that the regulation is clear and can be understood by persons who are required to comply with it.
6. An estimate has been made of the new reporting burdens or recordkeeping requirements necessary for compliance with the regulation.	The reporting burden and recordkeeping requirements have been reduced in the amended regulation. (See Value-Impact Statement.)
7. The name, address, and telephone number of a knowledgeable agency official is included in the publication.	The Federal Register notice promulgating the final rule contains the name, address, and telephone number of a knowledgeable agency official.
8. A plan for evaluating the regulation after its issuance has been developed.	Licensee and staff experience with the regulation will be used to evaluate the regulation. In addition, this regulation will be reviewed in the second cycle of NRC's periodic and systematic review process (1986-1991).

ENCLOSURE 4

ABSTRACT OF COMMENTS AND STAFF RESPONSE TO PROPOSED REVISION TO 10 CFR PART 50, APPENDICES G AND H, FRACTURE TOUGHNESS AND REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS, PUBLISHED FOR PUBLIC COMMENT IN THE FEDERAL REGISTER NOVEMBER 14, 1980.

Revised Draft
May 22, 1981

COMMENTERS ON PROPOSED REVISION TO
10 CFR PART 50, APPENDICES G AND H, AND
DATE THE COMMENT WAS DOCKETED

1.	C. W. Fay	Wisconsin Electric Power Company	1-8-81
2.	R. B. Bradbury	Stone and Webster Engineering Corporation	1-8-81
3.	A. E. Scherer	Combustion Engineering, Inc.	1-12-81
4.	T. M. Anderson	Westinghouse Electric Corp.	1-15-81
5.	J. S. Abel	Commonwealth Edison	1-16-81
6.	T. J. Sullivan	Consumers Power Co.	1-16-81
7.	G. G. Sherwood	General Electric Co.	1-19-81
8.	R. W. Jurgensen	American Electric Power Service Corporation	1-19-81
9.	C. M. Pratt	Power Authority of the State of New York	1-19-81
10.	J. H. Taylor	Babcock and Wilcox	1-23-81
11.	D. P. Hoffman	Consumers Power Company	2-5-81
12.	D. P. Hoffman	Consumers Power Company	2-5-81
13.	B. R. Silvia	Virginia Electric and Power Company	2-27-81

-- Each letter is numbered in the upper right hand corner.

-- Each comment in each letter is identified by a number in the left hand margin.

-- The attached resolution of public comments is keyed to refer first to the letter and then the comment within the letter (e.g., Comment 2 in the 3rd letter received would be referred to as Comment 3-2).

APPENDIX G, PARAGRAPH II.G. Comments 3-1 and 13-1

Comment 3-1 suggested that the phrase "as determined at the one-quarter thickness location" be inserted in the definition of beltline to insure "consistency with Regulatory Guide 1.99, Revision 1."

Response The suggestion was not accepted, primarily because it makes no difference in the selection of most limiting materials where the fluence values are estimated, provided a consistent thickness location is used. Regulatory Guide 1.99 does not define "beltline" or refer to the ¼ T position except in a different context. A footnote has been added to Paragraph V.B. that should provide a satisfactory response to Comment 3-1.

Comment 13-1 stated: "The revised definition of Beltline material is unclear in the statement, '...to be considered in the selection of the most limiting material.' To what extent, and with what tests is material in adjacent regions to be evaluated? Is it intended that surveillance specimens reflect such material?"

Response Yes, it is intended that surveillance specimens include material from "adjacent regions" (above or below the core) if the combined effect of reduced fluence but high radiation sensitivity (such as that caused by high copper content) makes that material controlling. Therefore, the test requirements for beltline material apply equally to material that directly surrounds the core and material in "adjacent regions" as described in paragraph II.G.

APPENDIX G, PARAGRAPH IV.A.1, Comment 7-1

Comment 7-1 asked if the Charpy upper-shelf energy values of 75 ft-lb and 50 ft-lb are average-of-three values or single Charpy specimen test results.

Response The upper-shelf energy requirements are average values. A footnote 1 has been added which refers the reader to ASTM E 185-79 for a definition of upper shelf energy. Footnote 1 is as follows:

"Defined in ASTM E 185-79 and -82, which are incorporated by reference in Appendix H."

APPENDIX G, PARAGRAPH IV.A.2. - Comments 1-1, 4-1, 5-1, 7-2, 9-1 and 12-1

Extensive comments were received from both PWR and BWR owners and vendors charging that the newly added requirement for a temperature of RT_{NDT} + 150°F at structural discontinuities at pressures exceeding 0.2 P_p (preoperational system hydrostatic test pressure) was overly restrictive. The quantity RT_{NDT} is the reference temperature of the highly-stressed material at the discontinuity. The critical locations in most cases are the fillets at the junctions of the closure flanges with the shell and head of the vessel.

The major problem for BWRs cited by commenters is the effect of the new requirement on the pressure-temperature limits for hydrostatic pressure tests and leak tests, not the limits for normal operation. Quoting from comment 5-1:

"Another concern with the 150°F margin above RT_{NDT} is that the system leakage and hydrostatic tests would be performed at a temperature closely approaching 212°F. The Technical Specifications require that primary containment integrity be maintained when the reactor water temperature is above 212°F. The proximity between the required 190°F metal temperature and the 212°F limit on water temperature could lead to station decisions to seal the drywell prior to pressure tests. This is not normally done and combined with the very slow heat-up rate above 150°F could add one or more critical path days to an outage."

Comments 4-1 and 9-1 asked that the 150°F be reduced to 50°F to be consistent with Branch Position MTEB 5-2, which reads as follows:

"Calculations need only be performed for the beltline region, if the assumed RT_{NDT} of the beltline is at least 50°F for all higher stressed regions."

Most of the comments focused on the RT_{NDT} + 150°F-at-0.2 P_p requirement, despite the alternative based on analysis, which was offered in the same paragraph:

> "...by showing that the margins of safety for those regions are equivalent to those required for the beltline when it is controlling."

Only comment 7-2 looked favorably on this alternative and recommended that the 150°F requirement be dropped in favor of increased quantitativeness concerning the flaw size to be assumed in the analysis.

Response: In response to the comments, paragraph IV.A.2 was changed to read as follows:

> 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[4] 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 shall 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 shall describe the methods of analysis used.

In response to the comments, plus a number of discussions at ASME Code working groups, paragraph IV.A.2 was revised in two respects. First, there are now separate requirements for hydrotest and normal operation. This was done to correct an oversight of the fact that margins of safety given in the ASME Code are lower for hydrotest than for normal operation. Second, the requirement for normal operation was reduced from RT_{NDT} + 150 to RT_{NDT} + 120 on the basis of several considerations as described below.

The closure flange regions are the structural discontinuities of principal concern, because boltup at ambient temperature produces high bending stress in the adjacent shell and head regions. Typical stress values from boltup alone are 40-50 ksi, not including the peak stresses at the fillets. Paragraph G2222(b) of the ASME Code requires that the bending stress be considered primary, consequently the ratio of K_{IR} to K_I should be 2 for the K_I value produced by the stress of 40-50 ksi acting on the postulated defect on the outer surface of the shell or head. Pressure stresses add to the boltup stresses in making the fracture mechanics calculation. At a pressure of 0.2 P_p, however, the stress addition is less than 5 ksi. At operating pressure, the beltline region is generally controlling.

Following the procedure given by the ASME Code, Appendix G, a fracture mechanics approach can be followed to derive the required temperatures, relative to RT_{NDT}, for hydrotest and for normal operation. To do so, the size of postulated defect must be chosen. The Code recommends a 1/4 T flaw for beltline calculations, but does not specify a depth for flaws at discontinuities. It recognizes that the assumed size for a nozzle region may be a fraction of that used for the beltline flaw and that justification for the difference must be made. The requirements given above--RT_{NDT} + 90 for hydrotest and RT_{NDT} + 120 for normal operation--is consistent with ASME Code, Appendix G, procedures and margins of safety and a postulated crack 0.6 in. deep by 3.6 in. long. This is approximately a 0.1T flaw for typical thicknesses of vessel heads, and is less than 0.1T for typical shell thicknesses.

There is some reason to believe that the flange areas are less likely to contain undetected large flaws than the beltline. During boltup, the stresses in the flange areas are higher than they are when pressurization begins, because there is some relaxation of bolt tension when the tensioning device is released and also when adjacent bolts are tightened. Thus, if propagation of a flaw is imminent, it should occur during boltup and the pop-in should be heard or sensed by the readings of bolt elongation.

A different approach to the determination of the required temperature margin at 0.2 P_p can be found in the following argument, which is based on the concept that boltup is analogous to a hydrotest performed at a low temperature, as far as the flange areas are concerned. The following argument does not rest on an assumed value of flaw size.

When boltup is completed, K_I for the largest flaw in the flange areas must less than K_{Ic} at the boltup temperature. Any flaw larger than the critical size has popped in and arrested at a value of K_{Ia}, which is lower than K_{Ic}. From the discussion given above, K_I at $0.2P_p$ will be essentially unchanged from that at boltup. Therefore, the basis for the temperature requirement at $0.2P_p$ is that K_{Ic} and also K_{IR} (to be consistent with the principles of the ASME Code, Appendix G) should be 2.0 times their values at the boltup temperature for normal operation and 1.5 times, for hydrotest and leak tests.

The temperature increment required to double K_{IR} (or to increase it by 1.5) is a function of the boltup temperature. The value chosen for hydrotest, RT_{NDT} + 90°F, produces a margin of 1.5 for a boltup temperature of RT_{NDT} + 38°F. The value chosen for normal operation, RT_{NDT} + 120F, produces a margin of 2.0 for a boltup temperature of RT_{NDT} + 40°F. For lower boltup temperatures, the margin of safety is greater. For higher boltup temperatures, the margin is less, but there is compensation in the reduced chance that the flanges would be cracked during boltup.

Boltup occurs at temperatures ranging from about 60°F to about 90°F, depending on the amount of residual heat in the core and on the ambient temperature. The value of RT_{NDT} for the material that is highly stressed by boltup is typically about 30°F, but in the absence of complete data is often assumed to be 60°F. Thus, boltup occurs in the range, RT_{NDT} to RT_{NDT} + 60°F. ASME Code rules limit the temperature to RT_{NDT}, minimum. The limit was lowered in 1977 from RT_{NDT} + 60°F to RT_{NDT}, for reasons of efficiency of operations and comfort of personnel doing the boltup.

The pressure, 0.2 P_p, above which the temperature requirements apply was chosen for operational reasons, and is consistent with ASME Code rules. With present practice, $0.2P_p$ is about 310 psig for BWRs and 625 psig for PWRs. Pump heat is used to warm the system, but in a PWR plant the pumps cannot be run until there is sufficient pressure to allow pump seals and bearings to function properly. That pressure is about 300-400 psig. In BWR plants, the pumps must run on the static head provided by the difference in elevation (unless the vessel is filled with water for a hydrotest). Inadvertent overpressurization is prevented by a low pressure set point on a power operated relief valve, for PWRs, and by the large vapor space in the reactor vessel, for BWRs.

The margin of safety is not impaired in the pressure range 0-to-0.2 P_p, because stress at the flange regions increases very little, if at all. The membrane stress increases only 5 ksi, and there is a reduction in bending stress as the pressure rises, hence the surface stress remains nearly constant.

Finally, it is necessary to explain why Branch Position MTEB 5-2 (which has a 50°F increment) is no longer considered adequate. The 50 degree increment was felt to provide a sufficient increase in K_{IR} to account for the increased stress at regions of structural discontinuity. However, since the Branch Position was written (Nov. 1975), the ASME Code allowable boltup temperature has been reduced from RT_{NDT} + 60°F to RT_{NDT}. At lower temperatures, the slope of the K_{IR} curve is flatter, i.e., the increment of temperature required to increase K_{IR} by a given factor is greater. Also, there is now more awareness of high stress levels at the closure flange regions, partly as a result of publication of work done for the ASME Section XI Working Group on Flaw Evaluation.*

<u>APPENDIX G, PARAGRAPH IV.A.4</u>, Comments 2-1, 4-3 and 13-2

Commenters questioned the meaning of "RT_{NDT}" in this paragraph. Does it apply to the reactor coolant pressure boundary or only to the reactor vessel? Does it apply to all areas of the reactor vessel or only to the beltline? And can it be applied throughout the lifetime of the vessel?

<u>Response</u> To clarify the meaning of paragraph IV.A.4, it has been revised to read as follows:

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

*Flaw Evaluation Procedures, EPRI NP-719-SR Special Report, August 1978, Prepared by ASME Section XI Working Group on Flaw Evaluation. Edited by T. U. Marston, Nuclear Power Division, EPRI.

APPENDIX G, PARAGRAPH V.B. Comment 1-2

Comment 1-2 asked for clarification or deletion of the sentence: "These predictions shall be made for the radiation condition at the tip of the assumed flaw at its deepest part."

Response The sentence in question has been revised to read as follows: "These predictions shall be made for the radiation conditions at the critical location on the crack front of the assumed flaw."[2]

In analyses that follow Appendix G of the ASME Code, the assumed flaw is a "¼ T" flaw (flaw depth equal to one fourth of the wall thickness). In many cases, but not always, the critical location is on the inside surface of the reactor vessel beltline, and the radiation conditions to be used in predicting damage are those at the ¼ T position. However, during heatup when thermal stresses are significant, the critical flaw may on the outside surface (the 3/4 T position). Or, the assumed flaw may not always be a ¼ T flaw. Or, if the stress gradient is large, the critical position along the crack front of a semielliptical surface crack may be near the surface, not at the deepest point. The wording was changed to reflect these possibilities, and footnote 2 was added to explain when the requirement refers to the ¼ T (or 3/4 T) location.

APPENDIX G, PARAGRAPH V.C. Comments 4-2 and 9-3

Commenters noted that the inspection interval is not specified, and urge that the interval given by Section XI of the ASME Code should be regarded as sufficient.

Response The purpose of paragraph V.C.1. is to make it clear that the Section XI inspection may not be adequate, either as to timing or to quality level, and the Director, Office of Nuclear Reactor Regulation, may require an examination to fit the circumstances of a specific case.

[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.

Comment 9-2 also suggested that paragraph V.C. be revised to require any one of the three steps called out in paragraphs 1, 2, and 3, not all three. The commenter stated that paragraph V.C. would require three analyses to be performed.

Response The analysis called for in paragraph V.C.3 must be based on the best estimate of flaw size in the material in question, and on the best evidence of radiation damage for that material. That is why volumetric examination and materials testing are specified as necessary additions to the analysis. Perhaps misunderstanding was caused by the phrase "...and any flaws evaluated according to Section XI..." in paragraph V.C.1. The phrase was intended to mean only that any indication found in the volumetric examination was to be characterized to determine the size, shape, orientation, location and nature of the flaw that produced the indication. The analysis of margin of safety for continued operation will not necessarily follow Section XI guidelines for flaw evaluation. To clear up the meaning, the word "evaluated" was changed to read "characterized" in the final rule.

APPENDIX H, PARAGRAPH II.A. Comments 1-3 and 13-4

Commenters called attention to a typographical error in the Federal Register. The parenthetical note, (E>1MeV) had read (E<1MeV). A correction was published in the Federal Register, page 77450.

APPENDIX H, PARAGRAPH II.B. Comment 7-3

Comment 7-3 objected to the requirements for number of capsules and withdrawal schedule that are given in Table 1 in ASTM E 185-79, which is incorporated by reference in Appendix H. For some BWRs, 4 surveillance capsules would be required instead of 3, the number required by Appendix H prior to these amendments. The change results from the fact that the breakpoint between 3 and 4 capsules is now given (in E 185-79) as a predicted Charpy shift of 100°F for the fluence condition at the vessel inside surface. Previously the criterion was not explicit, and it was interpreted to mean the fluence condition at the ¼ T position. Commenter argued that the fourth capsule adds cost and design hardship.

Response No change has been made in the regulation, for the following reasons. Hardship and extra cost of providing an extra capsule are neither large nor imminent. The rule applies only to vessels purchased to editions and addenda of the ASME Code issued after July 1979. Thus, it affects no plants now under construction. To effect a change in the requirements would mean that E 185-79 would have to be endorsed with an exception. The language of the exception would be somewhat involved, because the rules for number of capsules appear in the text and also in Table 1 of E 185-79. If the breakpoint between 3 or 4 capsules was changed, other changes would also be required. Continued use of the existing rules as given in Appendix H prior to these amendments is not acceptable, because the existing rules do not reflect our present judgment.

APPENDIX H, PARAGRAPH II.B.1 Comment 13-5

Comment 13-5, as explained by telephone conversations with the authors, was a result of lack of clarity in the effectivity requirements.

Response Paragraph II.B.1 has been reworded to clarify the requirements, particularly for the case of a capsule withdrawal between July 1979, when E 185-79 became effective, and the effective date of this revision of Appendices G and H. As revised, it reads as follows:

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. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after (insert the effective date of this rule), the test procedures and reporting requirements shall 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 (insert the effective date of this rule), either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used.

APPENDIX H, PARAGRAPH II.B.3 Comment 1-4

Comment 1-4 asked that a sentence be added to make it clear that the withdrawal schedule may be modified to coincide with refueling outages.

Response ASTM 185-79, which is incorporated by reference in Appendix H, contains the desired statement in paragraph 7.6.3.4: "Schedule the capsule withdrawals at the nearest vessel refueling date."

APPENDIX H, PARAGRAPHS III A AND III C

Comments 1-5, 3-2, 3-3, 4-4, 6-1, 6-2, 8-1, 10-1 and 11-1

Commenters suggested changes in the reporting requirements, especially the schedule. Several commenters (1-5, 3-1, 6-1 and 11-1) asked that the 30 day notice in advance of capsule withdrawal be deleted as unnecessary, because paragraph II.B.3. requires that the capsule withdrawal schedule be submitted for approval. Several commenters (1-5, 3-2, 4-4, 8-1, and 10-1) also asked for changes in the 90 day interval between completion of testing and submittal of the report to the NRC. Comment 10-1 suggested the addition of some flexibility for cases where there is good reason to take more time for evaluation of data. Comments 3-3 and 6-2 suggested that approval by the Director of NRR is not needed for the surveillance report, but only for the changes in pressure-temperature limits and any changes in operating procedures that are to be put in the Technical Specifications. Finally, Comment 6-2 suggested:

> "The proposed paragraph III C of Appendix H states that revised operating pressure-temperature limits and changes to operating procedures required to meet the revised limits must be submitted with the report of test results. It is recommended that these subjects not be addressed in the report but that the report should provide the expected date for submittal of the revised Techical Specifications which should be the proper document to address these subjects. The schedule for submittal of the

revised Technical Specifications should be related to the implementation date for the Technical Specifications and not the submittal of the test results."

Response In response to the comments, Section III. REPORT OF TESTS RESULTS is changed to read as follows:

A. Each capsule withdrawal and the test results shall 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 shall include the data required by ASTM E 185, as required by paragraph III.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 Technical Specifications is required, either in the operating pressure-temperature limits or in the procedures required to meet the limits, the expected date for submittal of the revised Technical Specifications shall be provided with the report.

The purpose of the schedule requirements in Paragraph III.A and III.C is to get the findings of the surveillance program reported as early as possible. If the results contain some technical surprise, that is all the more reason to avoid a delay in reporting. The basis for the schedule requirement was changed from the completion of testing to capsule withdrawal because the latter date was more easily defined, and because the purpose is to get surveillance information early, not to constrain the time spent on one part of the process. Comment 1-5 had suggested 1 year. After checking a number of surveillance reports, it appears that 1 year was on the low side of the range, but still feasible. When special problems require an extension of time, the request for extension prior to the end of the 1 year period will provide notification of the problem.

ENCLOSURE 5

DRAFT CONGRESSIONAL LETTER

Dear Mr. Chairman:

Enclosed for the information of the Subcommittee are copies of a Notice of Final Rulemaking to be published in the Federal Register.

The amendments of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," ~~comprises~~ comprise a general revision of Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," which have undergone only limited revision in over nine years of use. The purpose of the amendments is to update the requirements of Appendices G and H to be more consistent with current technology and pertinent National Standards. Some of the amendments are intended to clarify the applicability of these requirements to old and new plants. Two of the amendments modify requirements that have proved to be unduly conservative, and a number of other amendments shorten and simplify these regulations by replacing technical detail with references to appropriate National Standards.

Sincerely,

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosure: As stated

DOCKETED
USNRC

'94 MAR -9 P3:23

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY	)	Docket No. 50-440-OLA-3
	)	(Material Withdrawal Schedule)
(Perry Nuclear Power Plant, Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "NRC STAFF RESPONSE TO INTERVENORS' MOTION FOR SUMMARY DISPOSITION" and attachments thereto in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or as indicated by an asterisk through deposit in the Nuclear Regulatory Commission's internal mail system, this 7th day of March, 1994.

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Atomic Safety and Licensing
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