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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY	))	Docket No. 50-440-OLA-3 ASLBP No. 90-605-02-OLA
	)	
(Perry Nuclear Power Plant, Unit 1)	)))	(Material Withdrawal Schedule)

NRC STAFF'S BRIEF IN SUPPORT OF COMMISSION REVERSAL OF LBP-95-17

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April 26, 1996

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On March 7, 1996, the Commission issued an Order granting the petition filed by the Cleveland Electric Illuminating Company, *et al.* ("Licensees"), which had sought Commission review of the Licensing Board's "Memorandum and Order (Ruling on Motions for Summary Disposition)," granting summary disposition in favor of Intervenors Ohio Citizens for Responsible Energy, Inc. ("OCRE") and Ms. Susan L. Hiatt in this proceeding.[1] The Commission's Order directed the parties to file briefs on the issues under review, and further directed the parties to address "the significance for this case of 5 U.S.C. §§ 551(8) and (9) (defining 'license' and 'licensing')" (CLI-96-4, slip op. at 2). The NRC Staff ("Staff") hereby files its brief in this matter. For the reasons set forth below, the Staff submits that the Licensing Board's Memorandum and Order should be reversed.

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[1] *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Unit 1), CLI-96-4, 43 NRC __ (March 7, 1996), *granting review of* LBP-95-17, 42 NRC 137 (1995).

INTRODUCTION

On January 4, 1991, the NRC Staff issued Generic Letter (GL) 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," in which the Staff encouraged the removal of specimen capsule withdrawal schedules from licensee Technical Specifications (TS) as part of the Commission's line-item TS improvement program.[2] Shortly thereafter, on March 15, 1991, the Perry Licensees submitted an application to amend the Perry TS, whereby the Reactor Vessel Material Surveillance Program - Withdrawal Schedule (TS Table 4.4.6.1.3-1, pg. 3/4 4-22) would be relocated from the TS to the facility's Updated Safety Analysis Report (USAR), in accordance with GL 91-01.[3] A notice of opportunity for hearing on the Licensees' application was published in the *Federal Register* on July 24, 1991.[4]

On August 23, 1991, OCRE and Ms. Hiatt filed a timely "Petition for Leave to Intervene and Request for a Hearing" ("Petition"), in which they requested a hearing on the

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[2] The issuance of GL 91-01 is discussed *infra*, at 13-15.

[3] Letter from Michael D. Lyster to Document Control Desk, NRC, dated March 15, 1991. Prior to this request, the Perry TS, § 4.4.6.1.3, 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.

The Licensees proposed to delete the words italicized in the text above and to relocate the referenced Table from the TS to the USAR; the remainder of the TS would remain unchanged.

[4] "Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing," 56 Fed. Reg. 33950, 33961-62 (July 24, 1991).

proposal to remove the Withdrawal Schedule from the TS and relocate it in the USAR (Petition at 4-5). OCRE and Ms. Hiatt indicated that they agreed with the Licensees and Staff that the challenged portion of the proposed amendment "is purely an administrative matter which involves no significant hazards considerations" (*id.* at 5), but asserted that the removal of the schedule from the TS would deprive members of the public of their right to notice and opportunity for hearing on future changes to the schedule. On March 18, 1992, the Licensing Board denied the request for hearing, on the grounds that the Petitioners had failed to demonstrate injury-in-fact to an interest which may be affected by the proceeding and therefore failed to demonstrate standing to intervene. *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Unit 1), LBP-92-4, 35 NRC 114 (1992).

On September 30, 1993, the Commission reversed the Licensing Board's decision on standing, finding that the petitioners had shown a nexus between the asserted loss of procedural rights and their health and safety interests and had made a sufficient showing of their standing to intervene, subject to their submission of at least one admissible contention. *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Unit 1), CLI-93-21, 38 NRC 87, 93-96 (1993).[5] In its decision, the Commission further concluded that the Petitioners "should have an opportunity to raise and have resolved, subject to our rules of practice on the admission and litigation of contentions, whether the removal of the withdrawal schedule from the technical

[5] The Commission stated, "a fair reading of Petitioners' claims indicates that, at bottom, OCRE and Ms. Hiatt fear that if they are deprived of the opportunity to challenge future proposals to alter the withdrawal schedule, the surveillance of the Perry reactor vessel may become lax and prevent detection of a weakened reactor vessel, and ultimately result in an accidental release of radioactive fission products into the environment if the vessel should fail." CLI-93-21, *supra,* 38 NRC at 94. The Commission observed, "[t]he material condition of the plant's reactor vessel obviously bears on the health and safety of those members of the public who reside in the plant's vicinity." *Id.* at 96.

specifications is indeed an unlawful act." *Id.*, 38 NRC at 96. Accordingly, the Commission remanded the proceeding to the Licensing Board for consideration of OCRE and Ms. Hiatt's contention. *Id.*[6]

On November 12, 1993, OCRE and Ms. Hiatt filed a "Supplemental Petition for Leave to Intervene," which set forth a reformulated version of their contention, as follows:

> The portion of Amendment 45 to License No. NPF-58 which removed the reactor vessel material specimen withdrawal schedule from the plant Technical Specifications to the Updated Safety Analysis Report violates Section 189a of the Atomic Energy Act (42 USC 2239a) in that it deprives members of the public of the right to notice and opportunity for a hearing on any changes to the withdrawal schedule.[7]

On December 27, 1993, the Licensing Board issued an Order in which it, *inter alia*, admitted the Intervenors' contention;[8] and the parties then filed motions for summary disposition on the merits of the legal issue presented by the contention.

On October 4, 1995, the Licensing Board issued its Memorandum and Order ruling on the parties' motions for summary disposition. Therein, the Board concluded that, following the removal of the schedule from the Technical Specifications, notice and opportunity for hearing must be provided under section 189a of the Atomic Energy Act on any future changes to the

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[6] Subsequent to the Licensing Board's decision, but prior to the Commission's issuance of CLI-93-21, the Staff issued the requested amendment. *See* Letter from James R. Hall (NRC) to Michael D. Lyster, dated December 18, 1992, issuing Amendment No. 45 to the Perry operating license; and "Notice of Issuance of Amendment to Facility Operating License," 58 Fed. Reg. 5436, 5438 (Jan. 21, 1993).

[7] "Petitioners' Supplemental Petition for Leave to Intervene," dated November 12. 1993, at 1.

[8] "Order (Admitting Contention and Establishing Schedule)," dated December 27, 1993, at 2-3.

reactor vessel material specimen capsule withdrawal schedule, as long as 10 C.F.R. Part 50, Appendix H "remains in its current form." LBP-95-17, 42 NRC at 149. Significantly, however, while the Licensing Board opined that future changes to the withdrawal schedule must be treated as license amendments requiring notice and opportunity for hearing, it held that removal of the schedule from the TS is not an unlawful act, and it left intact the instant license amendment which had deleted the schedule from the TS. *Id.* at 141-42 and 148.

On November 7, 1995, the Licensees filed their "Petition for Review" ("Petition"), in which they urged the Commission to undertake a broad review of the Licensing Board's decision, arguing, in part, that the approval required by Appendix H was similar to the numerous approvals specified in other NRC regulations for which the Commission has not, heretofore, required license amendments (Petition at 5 n.7, and 10). The Intervenors filed an answer to the Petition on November 15, 1995, opposing Commission review; and the Staff filed an answer to the Petition on November 30, 1995, supporting a more limited review of the decision than had been requested by the Licensees.[9] On March 6, 1996, the Commission issued its Order granting review of LBP-95-17.

For the reasons set forth below, the Staff submits that the Commission should vacate and reverse the Licensing Board's decision, on the grounds that the Board erroneously concluded that 10 C.F.R. Part 50, Appendix H, requires all withdrawal schedule changes to be treated as

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[9] *See* "NRC Staff's Answer to Licensees' Petition for Commission Review," filed on November 30, 1995 ("Staff Answer"). In its Answer, the Staff stated its view that the Board's decision presents a "substantial question" concerning "a necessary legal conclusion [that] is "without governing precedent" and raises "a substantial and important question of law." The Staff further noted that the Commission has not previously rendered an opinion as to the proper interpretation of 10 C.F.R. Part 50, Appendix H, or as to the proper application of Generic Letter (GL) 91-01; that many nuclear power plant licensees have removed the withdrawal schedules from their TS in accordance with GL-91-01; and that the procedures for revising the schedules at these (and other) plants could be affected by the decision. Staff Answer at 2 n.1.

license amendments regardless of how insignificant those changes may be and regardless of whether the changes present an unreviewed safety question under 10 C.F.R. § 50.59.

STATEMENT OF ISSUES

I. Whether the Licensing Board Erroneously Concluded That 10 C.F.R. Part 50, Appendix H, Requires Prior Commission Approval of All Changes to An Approved Withdrawal Schedule.

II. Whether the Licensing Board Erroneously Concluded That Future Schedule Changes Must Be Treated As License Amendments, Regardless of How Insignificant the Changes May Be and Regardless of Whether the Changes Present an Unreviewed Safety Question Under 10 C.F.R. § 50.59.

III. Whether 5 U.S.C. §§ 551(8) and (9) (Defining "License" and "Licensing") Require That All Future Changes to the Withdrawal Schedule Be Treated As License Amendments.

ARGUMENT

Summary of Argument

The central issue in this proceeding is whether the Licensing Board correctly concluded that 10 C.F.R. Part 50, Appendix H, in its present form, requires that changes to an approved capsule withdrawal schedule be treated as license amendments subject to the provision of notice and an opportunity for hearing under section 189a of the Atomic Energy Act of 1954, as amended. The Licensing Board erred in its interpretation of Appendix H. Appendix H does not explicitly require that future schedule changes be treated as license amendments. Further, the regulatory history for this provision demonstrates that prior Commission approval and a license amendment application are not required for schedule changes that conform to the ASTM standard which is incorporated by reference in Appendix H. There is no requirement in the Commission's regulations, or in the Atomic Energy Act or the Administrative Procedure Act, that all such schedule changes be treated as license amendments. Moreover, even if Appendix H is construed to require prior Commission approval of all schedule changes, where those changes are in compliance with ASTM E 185, the agency's approval merely constitutes a *pro forma*, ministerial act for which notice and an opportunity for hearing are not required.

I. The Licensing Board Erroneously Concluded That 10 C.F.R. Part 50, Appendix H, Requires Prior Commission Approval of All Changes to An Approved Withdrawal Schedule.

An analysis of the issues presented in this matter must begin with an examination of 10 C.F.R. Part 50, Appendix H. In particular, § II.B.3 of Appendix H states as follows:

> 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.

The Licensing Board held that the "plain meaning" of this provision is that "the NRC must approve proposed schedules before they are implemented," including "any change to an already implemented schedule, significant or otherwise." LBP-95-17, 42 NRC at 145, 146. The Board concluded that the regulation is unambiguous on its face, and it therefore found it unnecessary to resort to the regulatory history of Appendix H to interpret the regulation. *Id.* at 145, 146.

The Licensing Board's determination as to the "plain meaning" of this provision is erroneous. While the regulation explicitly requires prior Commission approval of a "proposed schedule," it nowhere addresses the question of whether Commission approval of proposed changes to an already approved schedule is required -- *i.e.*, Appendix H is indeed ambiguous in this respect.[10] In order to resolve the question of whether prior Commission approval is

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[10] In fact, the Board, itself, appears to have found some ambiguity in the regulation, as demonstrated by its determination to go beyond the plain words of the regulation and to interpret the term "proposed schedule" in Appendix H to include, not just a proposed schedule, but also any proposed changes to a previously approved schedule. *See id.* at 146-48.

required for changes to a previously approved schedule, it is necessary to examine the regulatory history of Appendix H.

In its filings before the Licensing Board, the Staff set forth an extensive recitation of the regulatory history of Appendix H, which, it asserted, requires "prior approval" only of proposed changes that do not conform to the ASTM standard which is incorporated by reference in Appendix H.[11] While the Licensing Board initially determined that the rule is unambiguous and that it need not examine this history, it nonetheless did proceed to examine the regulatory history. It then concluded, however, that it did "not find the Staff's argument persuasive" (*Id.* at 146), and that, contrary to the Staff's view, the regulatory history "cannot be read reasonably to mean that only those proposed withdrawal schedules that do not conform to the applicable ASTM Code need be approved by the agency prior to implementation." *Id.* at 148. Accordingly, the Licensing Board rejected the Staff's historical interpretation and application of the rule (Staff Affidavit, ¶¶ 6, 14). *Id.*[12] The Board's determination in this regard was erroneous.

As indicated in the Introduction to Appendix H, the rule was developed to provide a means for obtaining test data that can be used in monitoring the effects of neutron irradiation

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[11] *See* "NRC Staff Response to Intervenors' Motion for Summary Disposition" ("Staff Response"), dated March 7, 1994, and "Affidavit of Barry J. Elliott, Christopher I. Grimes and Jack R. Strosnider" ("Staff Affidavit"), attached thereto.

[12] Significantly, the Licensing Board agreed with the Staff's view that "the 1983 amendment of Appendix H incorporated by reference the various editions of the E 185 ASTM Code (including Table 1 of those editions)." LBP-95-17, 42 NRC at 147. Table 1 of the ASTM standard, referred to by the Licensing Board, sets forth a withdrawal schedule and related criteria; a copy of ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)," was submitted as "Attachment 1" to the Staff's Response filed before the Licensing Board on March 7, 1994; a copy is also attached hereto for the convenience of the Commission and parties.

and the thermal environment on reactor vessel beltline materials, by the periodic withdrawal and examination of material specimens exposed in capsules to reactor vessel conditions. 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.
> >
> > * * *
> >
> > 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.

Thus, it is clear that the current version of Appendix H, § II.B.1, requires that a licensee's initial specimen program must comply with the applicable edition of ASTM E 185. 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.[13]

As originally formulated, Appendix H specified the number of capsules and specific withdrawal schedules to be followed by licensees, and described the circumstances under which modifications to those schedules would be appropriate.[14] Under these earlier formulations, the rule further provided that "proposed withdrawal schedules *that differ from those specified in paragraphs a. through f.*" were to be submitted to the Commission for approval, and stated that "[t]he proposed schedule shall not be implemented without prior Commission approval." 10 C.F.R. Part 50, Appendix H, § II.C.3.g (1980 ed.); emphasis added. *See* Affidavit ¶ 6.

In November 1980, the Commission published a proposed amendment to its fracture toughness and material surveillance program requirements, in which, *inter alia*, it proposed to restructure Appendix H and delete major portions thereof.[15] The Commission noted that "most" of former § II.C.3 -- which had contained the specific withdrawal schedules which licensees were required to follow -- would be deleted, "because the requirements for withdrawal schedules contained in the 1979 edition of ASTM E 185 provide satisfactory criteria for scheduling surveillance information gathering." Proposed Rule, 45 Fed. Reg. at 75537. Further, the Commission proposed to replace former paragraph II.C with new paragraph II.B; as pertinent

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[13] The Commission amended 10 C.F.R. § 50.60(b) in 1985, to clarify that alternatives to the requirements described in Appendices G and H may be used when an exemption is granted under 10 C.F.R. § 50.12. Final Rule, "Specific Exemptions; Clarification of Standards," 50 Fed. Reg. 50764, 50777 (Dec. 12, 1985).

[14] *See* Proposed Rule, "Fracture Toughness Requirements for Nuclear Power Reactors," 36 Fed. Reg. 12697, 12699 (July 3, 1971); *see also* Statement of Consideration, "Fracture Toughness and Surveillance Program Requirements," 38 Fed. Reg. 19012 (July 17, 1973).

[15] Proposed Rule, "Domestic Licensing of Production and Utilization Facilities; Fracture Toughness Requirements for Nuclear Power Reactors," 45 Fed. Reg. 75536 (Nov. 14, 1980).

here, new paragraph II.B.3 would replace former paragraph II.C.3.g. (recited *supra*, at 10), to read as follows:

> 3. Proposed withdrawal schedules shall be submitted with a technical justification therefor to the Director of Nuclear Reactor Regulation for approval. The proposed schedule shall not be implemented without prior approval.

Id., 45 Fed. Reg. at 75539.

In May 1983, the Commission adopted the proposed revisions to Appendix H in substantially similar form as the proposed rule.[16] In effect, the amendment deleted the withdrawal schedules from Appendix H, but retained the references to ASTM E 185. Significantly, the Commission incorporated ASTM E 185 by reference in Appendix H,[17] including the withdrawal schedule and criteria contained in ASTM E 185-79 and E 185-82.[18] Accordingly, since the withdrawal schedule and criteria for modifying the schedule, set forth in

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[16] *See* Statement of Consideration, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," 48 Fed. Reg. 24008 (May 27, 1983).

[17] *See* 10 C.F.R. Part 50, Appendix H, "Introduction" (noting that ASTM E 185-73, -79 and -82 were approved for incorporation by reference in Appendix H).

[18] While the withdrawal schedule and criteria contained in Table 1 of ASTM E 185-79 and ASTM E 185-82 (which contains criteria identical to those in ASTM E 185-79) are not referred to specifically in the regulation, the Commission indicated its intent to incorporate those matters in its 1983 revisions of the rule. For instance, the value/impact statement prepared in conjunction with the proposed rule, and the regulatory analysis prepared in conjunction with the final rule, both state that "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." *See* SECY-83-80, "10 C.F.R. Part 50 -- General Revision of Appendices G and H, Fracture Toughness and Reactor Vessel Material Surveillance Requirements," Feb. 25, 1983, Enclosure 2 (Regulatory Analysis), at 1 (referred to in 48 Fed. Reg. at 24008); and SECY-80-375, Enclosure 2 (Value/Impact Statement), at 1 (referred to in 45 Fed. Reg. at 75537); *see also* SECY-83-80, *supra,* Enclosure 4 (Abstract of Comments and Staff Response), Response to Comment 7-3, at 10-11 (accepting the commenter's view, without disagreement, that "Table 1 in ASTM E 19 85-79 . . . is incorporated by reference in Appendix H"). A copy of SECY-83-80, together with its Enclosures, is attached to the Staff's Response filed before the Licensing Board on March 7, 1994, as "Attachment 2."

ASTM E 185-79 and ASTM E 185-82, are incorporated by reference in the rule, the Commission deleted the withdrawal schedule which previously had been contained in Appendix H -- with the expectation that licensees would follow the ASTM standard or obtain prior Commission approval for deviations. *See* Affidavit ¶ 6.

As noted *supra*, at 8 n.12, the Licensing Board agreed with the Staff's view that ASTM E 185 (including Table 1 thereto) was incorporated by reference in Appendix H. *See* LBP-95-17, 42 NRC at 147. The Board disagreed, however, with the Staff's reading of the regulatory history as requiring prior approval only for schedule changes which do not conform to the schedule in ASTM E 185 -- based on the Board's observation that in 1983, when the Commission "deleted the provision that specifically limited any requirement for prior agency approval of schedules only to those that differed from the schedules set forth in the regulation," it simultaneously "substituted a new comprehensive requirement that the agency approve *all* proposed schedules prior to implementation." *Id.* at 147-48 (emphasis in original).

The Board's conclusions in this regard are inherently inconsistent. Having incorporated the ASTM standard into the regulation, the withdrawal schedule set forth therein was, *ipso facto*, already approved for implementation. If, as the Board agreed, the schedule set forth in the ASTM Code is incorporated by reference in the rule, what reason would there have been for the Commission to simultaneously require that "*all* proposed schedules" be approved in advance by the Commission? The only rational reading of the Commission's 1983 amendment, therefore, in light of the rule's incorporation of the ASTM schedule, is either (a) that prior approval is required only for schedule changes that do not conform to the ASTM standard, or (b) that Commission approval of all schedule changes <u>is</u> required -- but only to verify that the changes are consistent with the ASTM standard; *i.e.*, while schedule changes that are consistent with the ASTM standard were already deemed to be acceptable and need not be further specifically approved in advance, schedule changes that are not consistent with the ASTM standard do require specific approval prior to implementation.[19] Further, where the Commission's "approval" of a proposed schedule change constitutes merely a verification that the change conforms to the ASTM standard incorporated in the regulation, such approval merely constitutes a *pro forma,* ministerial determination -- which, as discussed *infra* at 21-22, would not require adjudicatory hearings and a license amendment.

II. The Licensing Board Erred in Concluding That Future Schedule Changes Must Be Treated As License Amendments, Regardless of How Insignificant the Changes May Be or Whether the Changes Present an Unreviewed Safety Question Under 10 C.F.R. § 50.59.

On January 4, 1991, as part of the Commission's line-item TS improvement program designed to eliminate unnecessary TS requirements for nuclear power reactors, the Staff issued GL 91-01 ("Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications"). Therein, the Staff indicated it would approve the removal of withdrawal schedules from the TS, subject to a requirement that licensees doing so commit to

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[19] Following the deletion of the schedule from Appendix H in the 1983 rule change, the Staff concluded that proposed schedule changes which are in conformance with ASTM E 185-79 (and ASTM E 185-82) already satisfy the requirements of Appendix H, and do not require specific prior approval (Affidavit ¶¶ 6, 12, and 14); only those changes which do not conform to the referenced ASTM standard required specific approval prior to implementation. Accordingly, under the Staff's practice subsequent to the 1983 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 the schedule was then located in a licensee's TS, and any modification to the TS required a license amendment under 10 C.F.R. § 50.59(c). *See* Affidavit, ¶ 6.

include the schedules in the next revision of their Updated Safety Analysis Reports (USARs). Affidavit ¶ 12.[20] The Staff stated:

> 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).[21]

The Staff's conclusion that the withdrawal schedule could be removed from the TS was based on the Staff's determination that inclusion of the withdrawal schedule in the TS was not specifically required by § 182a of the Atomic Energy Act of 1954, as amended, or by 10 C.F.R. § 50.36 or other regulations, and was not necessary since Appendix H provides an adequate means of controlling proposed changes to withdrawal schedules. Affidavit, ¶ 9.

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[20] As part of the line-item TS improvement program, potential 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. *See* Affidavit, ¶ 8.

[21] In its filings before the Licensing Board, the Staff indicated that, "in hindsight, it appears that GL 91-01 does not express the Staff's views on this matter with precision." Staff Response at 27 n.33. Indeed, the Staff recognizes, as the Board noted (LBP-95-17, 42 NRC at 148 n.23), that GL 91-01 contains language which is inconsistent with the views expressed by the Staff in this proceeding. Thus, GL 91-01 states (Enclosure, at 1) that § II.B.3. of Appendix H "mandate[s] prior NRC approval of any changes to the withdrawal schedule," that "placement of this schedule in the TS duplicates the controls on changes to this schedule that have been established by Appendix H," and that "this duplication is unnecessary." Similarly, the *Federal Register* notice of the instant license amendment, removing the withdrawal schedule from the Perry TS, stated: "The relocation of the surveillance capsule withdrawal schedule from the TS to the USAR in accordance with GL 91-01, is a purely administrative change; NRC prior approval is still necessary for any change to the schedule itself." 56 Fed. Reg. at 33962.

Further, the Staff had determined that the schedule was not of "controlling importance to safety" and did not require rigid "conditions of operation which cannot be changed without prior Commission approval" -- *i.e.*, it did not constitute a matter which is "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." *Id.*, ¶ 11.[22]

Significantly, the Intervenors and Licensing Board both agreed with the Staff's view that there is no statutory or regulatory requirement for specimen capsule withdrawal schedules to be included in a facility's Technical Specifications -- and the Board held that the withdrawal schedule may lawfully be removed from the TS. 42 NRC at 141-42, 148.[23] However, the Licensing Board then erroneously agreed with the Intervenors' assertion that all future changes to the schedule constitute material "licensing actions," so as to require a license amendment and notice and opportunity for a hearing under § 189a of the Atomic Energy Act. *Id.* at 148-49.

The Licensing Board's determination in this regard is erroneous, in that Appendix H does not require specific prior approval of all schedule changes -- and even if it does, such approval would merely constitute a *pro forma* verification that the change is consistent with the ASTM standard. Further, the Licensing Board failed to recognize that under the Commission's

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[22] This determination was consistent with the guidelines in the Commission's interim and final policy statements on TS improvements. *See* "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," 52 Fed. Reg. 3788 (Feb. 6, 1987); and "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors", 58 Fed. Reg. 39132, 39133-34, 39136 (July 22, 1993). The Final Policy Statement provided four criteria for determining which items are to be included in a facility's TS (58 Fed. Reg. at 39137-38), which were later adopted, in substantially the same form, in recent revisions to 10 C.F.R. § 50.36. *See* 10 C.F.R. § 50.36(c)(2), *as amended*, 60 Fed. Reg. 36953 (July 19, 1995).

[23] *See also* "Intervenors' Answer to Licensees' Petition for Review," dated November 15, 1995, at 2, 7; and "Motion for Summary Disposition, dated February 7, 1994, at 6, 7.

regulations in 10 C.F.R. § 50.59, a licensee is authorized to make numerous changes to its manner of operation without specific prior approval by the NRC, as long as the changes do not involve a change to the TS and do not present an unreviewed safety question, as stated in 10 C.F.R. § 50.59(c).[24] Thus, under 10 C.F.R § 50.59, schedule changes that are consistent with ASTM E 185-79 or -82 would not involve an unreviewed safety question and could be effectuated without specific Commission approval or a license amendment; but proposed schedule changes that are not consistent with ASTM E 185-79 or -82 would likely be deemed to involve an unreviewed safety question and would require prior NRC approval and a license amendment, under 10 C.F.R. § 50.59(c). *See* Affidavit, ¶ 14 (explaining which schedule changes the Staff would likely deem to require a license amendment).[25]

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[24] *See* 10 C.F.R. § 50.59(a)(1) (changes to a facility or to the procedures described in the FSAR may be made without prior NRC approval, unless the change involves an unreviewed safety question or a change in the TS). The regulatory scheme embodied in 10 C.F.R. §§ 50.59 was discussed, most recently, in *Citizens Awareness Network v. NRC*, 59 F.3d 284, 287 (1st Cir. 1995), where the Court described this regulatory framework as "allowing a licensee to modify its facilities without NRC supervision, unless the modification is inconsistent with the license or involves an 'unreviewed safety question.'" As the Court further observed (*Id.*):

> If the proposed change is inconsistent with the license, or does involve an unreviewed safety question (as that term is defined in 10 C.F.R. § 50.59(a)(2)(ii)), the licensee must apply to the Commission for a license amendment, 10 C.F.R. § 50.59(c), and only then are the statutory hearing rights of § 189a triggered.

[25] This is consistent with the Final Policy Statement, where the Commission concluded that requirements which do not require prior Staff approval should be relocated from the TS to other documents (such as the FSAR) and controlled by more appropriate means, such as through the use of 10 C.F.R. § 50.59, and enforcement action to assure compliance therewith. 58 Fed. Reg. at 39134, 39138. *Accord, Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273-74 (1979) (matters "which have not been found to possess safety implications of sufficient gravity and immediacy to warrant their translation into technical specifications" are adequately addressed by the reporting requirements in 10 C.F.R. § 50.59, and the Staff is thereby in a position to monitor both facility changes and licensee adherence to FSAR commitments and to take any remedial action that may be appropriate).

Contrary to the Licensing Board's conclusion, even if Appendix H is found to require Commission approval of changes to a licensee's withdrawal schedule, those changes would not necessarily constitute a "material" licensing action. Rather, as discussed *infra* at 20-22, changes which are consistent with the ASTM standard incorporated in Appendix H do not involve an unreviewed safety question, and therefore do not require a license amendment under 10 C.F.R. § 50.59. Commission approval of such changes would merely constitute a mechanical verification that the schedule conforms to the ASTM standard incorporated in Appendix H.[26] A "material" licensing action would only be associated with schedule changes that are not consistent with that ASTM standard (*see* Affidavit, ¶¶ 11, 16 (question (b)).[27] For such "material" licensing actions, license amendments would likely be required (*Id.*, ¶ 14), preceded by the publication of notice and an opportunity for hearing.[28] No more is required under the Atomic Energy Act. Accordingly, the Licensing Board's reliance on *Union of Concerned*

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[26] ASTM E 185 was established as the governing standard in 1983 upon its incorporation by reference in Appendix H, at which time the Commission provided notice and opportunity for public comment in the rulemaking proceeding. *See* 45 Fed. Reg. at 75536, 75537.

[27] In addition, a material licensing action could involve the approval of changes in the limiting conditions for operation (LCOs) governing the related pressure/temperature (P-T) limits -- which would be brought about by changes in material specimen test <u>results</u> rather than changes in the withdrawal schedule. *See* Affidavit, ¶¶ 11, 16 (question (b)).

[28] This conclusion is consistent with the Commission's decision in CLI-93-21, that the Intervenors may raise, in this proceeding, the question of whether removal of the withdrawal schedule from the TS would deprive them of the opportunity to challenge future changes to the schedule which might result in a "weakened reactor vessel" and "an accidental release of radioactive fission products into the environment if the vessel should fail." CLI-93-21, 38 NRC at 94. The Intervenors would have the opportunity to challenge any significant proposed schedule change in the future, through the license amendment proceeding which would be held on changes which are not consistent with the ASTM standard. *See* Affidavit, ¶ 14 (proposed changes that are not consistent with ASTM E 185-79 or -82 "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)").

Scientists v. NRC, 735 F.2d 1437, 1451 (D.C. Cir. 1984), and *Citizens Awareness Network v. NRC*, 59 F.3d 284, 294 (1st Cir. 1995), in concluding that any change to the withdrawal schedule requires a license amendment and notice and opportunity for hearing (LBP-95-17, 42 NRC at 148-49), is in error.

III. The Provisions of 5 U.S.C. §§ 551(8) and (9) (Defining "License" and "Licensing") Do Not Require That All Future Changes to the Withdrawal Schedule Be Treated As License Amendments.

In its Order granting review of this matter, the Commission directed the parties to address the question of "the significance for this case of 5 U.S.C. §§ 551(8) and (9) (defining 'license' and 'licensing')." CLI-96-4, at 2, referring to the Administrative Procedure Act (APA). Those provisions state as follows:

> **Sec. 551. Definitions**
>
> For the purpose of this subchapter -
>
> (8) "license" includes the whole or a part of an agency permit, certificate, approval, registration, charter, membership, statutory exemption or other form of permission;
>
> (9) "licensing" includes agency process respecting the grant, renewal, denial, revocation, suspension, annulment, withdrawal, limitation, amendment, modification, or conditioning of a license . . . .

The significance of these provisions for licensing actions, in general, is that pursuant to 5 U.S.C. § 558(c), certain hearing rights may pertain to licensing actions -- where hearings are otherwise required by the agency's governing statute or other applicable law.[29] However, neither § 551 of the APA, nor any other provision in that statute, requires that hearings be held on a licensee's application to amend its license or on an agency's approval of such an application, except to the extent that such hearings are required to be held under the Atomic Energy Act of 1954, as amended.[30]

In this regard, § 189a of the Atomic Energy Act provides, in pertinent part:

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[29] Section 558(c) provides in pertinent part:

> (c) When application is made for a license required by law, the agency, with due regard for the rights and privileges of all the interested parties or adversely affected persons . . . shall set and complete proceedings required to be conducted in accordance with section 556 and 557 of this title or other proceedings required by law and shall make its decision.

Sections 556 and 557 of the APA set forth certain procedural requirements that pertain when a hearing is held under § 556 of the APA; § 556, in turn, applies to hearings held under § 553 (rulemaking) or § 554 (adjudications). *Id.*, § 556(a). With respect to adjudicatory hearings, § 554 provides, in pertinent part, as follows:

> (a) This section applies, according to the provisions thereof, in every case of adjudication required by statute to be determined on the record after opportunity for an agency hearing, except to the extent that there is involved . . .
>
> * * *
>
> (3) proceedings in which decisions rest solely on inspections, tests, or elections . . . .

[30] *See, e.g., Three Mile Alert, Inc. v. NRC*, 771 F.2d 720 (3rd Cir. 1985), *cert. denied*, 475 U.S. 1082 (1986); *Gallagher & Ascher Co. v. Simon*, 687 F.2d 1067, 1072 (7th Cir. 1982); *Seacoast Anti-Pollution League v. Costle*, 572 F.2d 872, 878 n.11 (1st Cir.), *cert. denied*, 439 U.S. 824 (1978); *Advanced Medical Systems, Inc.* (One Factory Row, Geneva, OH 44041), ALAB-929, 31 NRC 271, 282, 285-86 (1990).

Sec. 189. Hearings and Judicial Review.

> a(1) *In any proceeding under this Act, for the* granting, suspending, revoking, or *amending of any license* or construction permit . . . the Commission shall grant a hearing upon the request of any person whose interest may be affected by the proceeding, and shall admit any such person as a party to the proceeding. . . .

While section 189a of the Act provides an opportunity for hearing on operating license amendments, it is clear that not every change to a licensee's manner of operation constitutes a license amendment for which a hearing is required under the Act. To the contrary, as discussed *supra* at 15-17, the Commission's regulations provide authority for NRC licensees to modify their manner of operation, without prior Commission approval, under the provisions of 10 C.F.R. § 50.59. Only where the change involves a change to a technical specification, or a change to the facility or procedures which involve an unreviewed safety question, would prior Commission approval, and a license amendment, be required. *Id.* Similarly, only where a proposed change to a withdrawal schedule does not conform to ASTM 185 E (incorporated by reference in Appendix H), would prior Commission approval and a license amendment, with its attendant notice and opportunity for hearing, be required. This regulatory scheme does not contravene section 189a of the Atomic Energy Act. *See Union of Concerned Scientists v. NRC*, 735 F.2d 1437, 1443, 1447 (D.C. Cir. 1984), *cert denied,* 469 U.S. 1132 (1985) (requiring hearings on matters that are "material" to a Commission licensing decision); and *Sholly v. NRC*, 651 F.2d 780, 791 (D.C. Cir. 1980), *vacated on other grounds*, 459 U.S. 1194 (1983) (holding that an action which grants a licensee authority to do something it otherwise could not do under the existing license is a license amendment).

The APA's definition of the terms "license" and "licensing" has been recognized to be very broad.[31] However, cases interpreting these terms indicate that a license amendment, with its attendant notice and opportunity for hearing, is not required in every instance that the Commission approves (or fails to disapprove)[32] a licensee's action.

In this regard, the decision of the Second Circuit Court of Appeals, declining to require hearings in a similar situation, is instructive. The Court stated:

> [T]hough the "approval" procedure may appear on the surface to fall within the broad definition of "licensing" under the Administrative Procedure Act, 5 U.S.C. §§ 551(8),(9), in reality the procedure involves the performance by DOT of a skilled, but essentially pro forma act -- *i.e.*, determining whether, on the face of an application, [an applicant] has shown the capability for meeting [regulatory] specifications . . . .

American Cylinder Manufacturers' Committee v. Department of Transportation, 578 F.2d 24, 27 (2nd Cir. 1978).[33]

The Commission approved a similar approach with regard to inspections, tests and acceptance criteria, upon adopting the "combined license" regulations in 10 C.F.R. Part 52. There, the Commission concluded that "findings which rest solely on the results of tests and

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[31] *See, e.g.*, *Air North America v. Department of Transportation*, 937 F. 2d 1427, 1437 (9th Cir. 1991); *Seacoast Anti-Pollution League v. Costle*, 572 F.2d 872, 880 n.15 (1st Cir.), *cert. denied*, 439 U.S. 824 (1978).

[32] *See, e.g.*, *Sheridan Kalorama Historical Ass'n v. Christopher*, 49 F.3d 750, 756 (D.C. Cir. 1995) (Secretary of State's failure to disapprove a proposal did not render it a federally licensed undertaking, even though § 551(8) of the APA broadly defines "license" to include "any form of permission").

[33] *Cf.* 5 U.S.C. § 554(a)(3) (excepting from the APA's procedural requirements those adjudicatory "proceedings in which decisions rest solely on inspections, tests, or elections"). *See also Air North America v. Department of Transportation*, 937 F.2d 1427, 1438 (9th Cir. 1991) (hearings were not required under 5 U.S.C. § 551(8) where no factual issues were in dispute); *Atlantic Richfield Co. v. United States*, 774 F.2d 1193, 1203 (D.C. Cir. 1985) (same).

inspections should not be adjudicated," and indicated that it disfavored hearings in "cut-and-dried" proceedings involving "highly detailed 'objective criteria' entailing little judgment and discretion in their application, and not involving questions of 'credibility, conflicts, and sufficiency.'"[34]

The ASTM standard, which was incorporated by reference in Appendix H after notice and opportunity for comment were provided in conformance with the APA and Atomic Energy Act, constitute the same type of objective acceptance criteria as were present in *American Cylinder Manufacturers* and the Commission's 10 C.F.R. Part 52 rulemaking. Thus, even if the Licensing Board correctly concluded that Appendix H requires prior approval of all changes to an approved withdrawal schedule, adjudicatory hearings would not be required on schedule changes that conform to the ASTM standard incorporated by reference in the rule, since the agency's approval would merely constitute a *pro forma* determination that the change is consistent with ASTM E 185 and Table 1 thereto.

Finally, the Staff notes that the Licensees, in their Petition, asserted that the Licensing Board's decision has broad and far-reaching potential consequences, in that it may lead to the treatment of numerous approvals required under NRC regulations as license amendments (Petition at 1, 2). The Staff agrees that the Board's decision could be read broadly, and that numerous "approvals" are specified in the regulations. However, the actual effect of the Board's decision is quite limited: The only direct effect of the decision is on Appendix H "approvals," in that henceforth, after a withdrawal schedule has been removed from the TS, changes to that

[34] Statement of Consideration, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors," 54 Fed. Reg. 15372, 15380 (April 18, 1989), *citing UCS* and the hearing exception specified in 5 U.S.C. § 554(a)(3); *see also UCS, supra*, 735 F.2d at 1451 (limiting application of the exception in APA § 554(a)(3) to situations involving pre-established objective acceptance criteria).

schedule are to be made by license amendment, for which notice and opportunity for hearing are to be provided -- just as was done before GL-91-01 authorized the removal of these schedules from the TS. Accordingly, the Staff believes that the Commission need not undertake, in this proceeding, a broad review of regulations other than Appendix H as was proposed by the Licensees.[35]

CONCLUSION

For the reasons set forth above, the Staff recommends that the Commission should vacate and reverse the Licensing Board's decision in LBP-95-17.

Respectfully submitted,

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Sherwin E. Turk
Counsel for NRC Staff

Dated at Rockville, Maryland
this 26th day of April, 1996

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[35] Nonetheless, the Staff believes that the Commission should consider the impact of the Board's decision on regulations other than Appendix H. In the Staff's Answer to the Licensees' Petition, the Staff indicated its view that the Commission (or Staff) could "undertake to consider the broader implications of the Licensing Board's decision, with respect to the numerous regulatory requirements for NRC approval of licensee submittals," but suggested "that any such broad consideration should be undertaken on a generic basis, outside the scope of this adjudicatory proceeding." *Id.* at 10. Further, the Staff indicated its view that "if a broad review is undertaken by the Commission, any interim guidance which the Commission may provide during the pendency of that review would undoubtedly be of benefit to the Staff, licensees and other interested persons." The Staff reiterates its views in this regard.

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.

$^{\epsilon 1}$ Note—Section 9.2.3 was corrected editorially and the designation date was changed July 1, 1982.
$^{\epsilon 2}$ Note—The title was changed editorially in July 1983.

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/m^2 (1×10^{17} n/cm^2) 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 ~~mid-thickness of the plates shall not be used~~ 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[8] 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.

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

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 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.

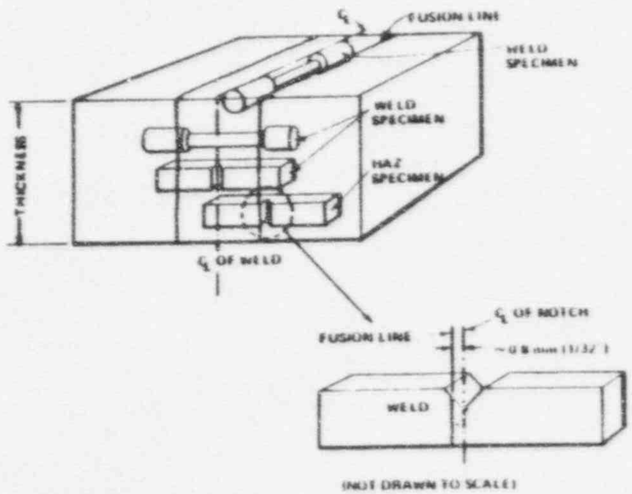

FIG. 1 Location of Test Specimens Within Weld and Heat-Affected-Zone (HAZ) Test Material

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NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	Docket No. 50-440-OLA-3
THE CLEVELAND ELECTRIC	)	ASLBP No. 90-605-02-OLA
ILLUMINATING COMPANY	)	
	)	
(Perry Nuclear Power Plant,	)	(Material Withdrawal Schedule)
Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S BRIEF IN SUPPORT OF COMMISSION REVERSAL OF LBP-95-17" in the above-captioned proceedings have been served on the following, by deposit in the United States mail or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 26th day of April 1996.

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