SUPPLEMENTAL SAFETY EVALUATION

SEISMIC QUALIFICATION OF THE AUXILIARY FEEDWATER SYSTEM (MPA C-14)

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

By memorandum from L. S. Rubenstein to Gus Lainas dated October 22, 1985, the staff transmitted its safety evaluation report (SER) concerning compliance with the criteria of Generic Letter 81-14, "Seismic Qualification of Auxiliary Feedwater Systems" for Oconee, Units 1, 2 and 3. Compliance with the criteria of the generic letter was to be demonstrated in order to assure that the requirements of GDC 2 and 34 were satisfied for assuring post seismic event shutdown decay heat removal capability. The staff SER identified three open items as follows:

1. The capability of the auxiliary feedwater (AFW) system and/or safe shutdown facility to withstand a safe shutdown earthquake (SSE) concurrent with a single active failure.

2. Requirements for the isolation boundary between seismic and nonseismic portions of the AFW system.

3. Walkdown of the currently nonseismically qualified areas of the AFW system.

The following is our evaluation of these issues.

1. <u>AFW System Single Failure Capability Following An SSE</u>

Generic Letter 81-14 states that licensees were to demonstrate that the AFW system could perform its shutdown decay heat removal safety function following an SSE [maximum hypothetical earthquake (MHE)for Oconee] and concurrent single active failure. Alternatively, the licensee could demonstrate the availability of a seismically qualified alternative system for performing this function. In the SER, the staff noted that the AFW system in each of the three Oconee units was located at the basement elevation of the turbine building and was, therefore, subject to a complete

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failure as a result of flooding caused by rupture of the nonseismic condenser circulating water line. In such an event, the only identified means for shutdown decay heat removal for the three units would be the seismically qualified safe shutdown facility (SSF) auxiliary service water (ASW) pump. However, because the SSF consists of a single ASW pump for supplying feedwater flow to all three units, a single failure in it results in a loss of decay heat removal capability through the steam generators. Consequently, we informed the licensee that we were pursuing a possible backfit of Oconee to correct this condition and satisfy the requirements of GDC 2 and 34 for decay heat removal capability following an earthquake.

By letters dated April 28 and May 7, 1986, the licensee provided additional information regarding this concern in order to support their assertion that previously completed modifications will assure adequate post-seismic decay heat removal capability and no further backfit is cost-beneficial. The licensee indicated that penetration seals and waterproof doors have been installed between the turbine building and the auxiliary building in each unit to provide waterproofing to a height of 20 feet above the turbine building basement floor. Thus, the high pressure injection (HPI) system, low pressure injection (LPI) system, auxiliary service water system, and reactor building spray system pumps located in the auxiliary building would be available as an alternative to the AFW system and SSF ASW pump for shutdown decay heat removal. Further, the licensee indicated that revised operating procedures have improved the operator's ability to quickly respond to a turbine building flood by providing guidance on means to isolate the circulating water system, initiating feed-and-bleed utilizing the HPI pumps and starting the SSF ASW pump.

The staff performed a quantitative probabilistic evaluation of the above information contained in the Oconee PRA. Based on this review, a core melt frequency and an associated cost benefit were determined for the seismic flooding scenario accounting for the indicated plant improvements.

The results of this analysis indicated that adequate core melt protection has been provided and no further plant improvements to correct seismic flooding concerns are warranted. This determination is based on the flooding protection provided for the HPI and LPI pumps for use in the feed-and-bleed mode along with the SSF ASW pump which serve as a suitable redundant alternative decay heat removal means to the AFW system since the AFW system itself is unprotected from flooding and, therefore, assumed unavailable following an SSE. The staff, therefore, concludes that the concern regarding post-seismic event decay heat removal capability and concurrent single failure is resolved, and the criteria of Generic Letter 81-14 regarding seismically qualified alternative decay heat removal means is satisfied.

2. Isolation Between Seismic and Nonseismic Portions of the AFW System

By letters dated February 6, 1986 and March 5, 1986 (two letters), the licensee indicated that as a result of their continuing review of the seismic qualification of the AFW system in response to staff concerns, a condition was identified outside the design basis for Oconee. Specifically, a) certain manually operated boundary valves are not normally closed, b) certain valves do not have complete seismic qualification documentation, and c) some piping attached to the upper surge tanks is not seismically qualified. In the above letters, the licensee provided a safety evaluation which discussed the capability to safely shutdown the plant in the event of an SSE (MHE for Oconee) given the specifically indicated deficiencies and described proposed corrective actions for assuring AFW system seismic qualification in each case. The licensee also identified a schedule for implementation of required modifications in order to achieve AFW system seismic qualification in accordance with the design basis.

The licensee's safety evaluation provided a justification for continued operation based on the inherent seismic resistance of nonseismically qualified piping and valves, and on the diversity of seismically qualified alternative means of decay heat removal. The licensee indicated that the results of the study of earthquake effects on power plants being performed by the Seismic Qualification Utilities Group (SQUG) have shown nonseismically qualified piping and valves to generally remain functional in seismic events. This capability has been evaluated in depth by the staff in the resolution of USI A-46, Seismic Qualification of Equipment in Operating Nuclear Plants. More importantly however, as discussed in Item 1, above, the seismically qualified SSF ASW pump and feed-and-bleed capability are available for decay heat removal should the AFW system fail following an SSE. These additional means for assuring shutdown are not only significant in the interim while the identified AFW system seismic deficiencies are corrected, but also serve as additional defense-in-depth protection against core melt in the long term given the seismically-induced flooding vulnerability of the AFW system discussed previously.

Corrective actions identified by the licensee for AFW system seismic qualification deficiencies will be one of the following as appropriate:

a) Normally open boundary valves will be closed, or will be modified to be remotely operated, or analysis will demonstrate that failure of piping beyond these valves will have no impact on system function.

b) Seismically unqualified piping will be analyzed and supported to withstand an MHE.

c) Seismically unqualified valves will be shown capable of withstanding an MHE, or will be replaced, or analysis will demonstrate that failure will have no impact on system function.

The staff finds the above identified corrective actions to be in accordance with the AFW system design basis and, therefore, acceptable for assuring its seismic qualification.

The licensee's letter of March 5, 1986 stated that a schedule for implementation of the required modifications will be provided to the staff by January 5, 1987, with completion of the modifications estimated to be accomplished by January 1990. In the interim, the licensee indicated that plant procedures have been revised to instruct the operator to investigate those locations where normally open valves exist in interfaces between seismic and nonseismic portions of the AFW system following an SSE in order that any necessary action can be taken to isolate the boundary. Because of the above indicated alternative decay heat removal means, the staff concurs with the licensee's proposed schedule for implementation of corrective actions, however, any schedule slippage should be properly justified. We further conclude that adequate post-seismic event shutdown decay heat removal capability is provided for assuring continued plant safety, and the concern regarding isolation of the seismic/nonseismic boundary is resolved.

3. Walkdown of Nonseismically Qualified Areas of the AFW System

As indicated in the licensee's March 5, 1986 letter which contained LER 269/86-02, the licensee has reviewed the seismic/nonseismic interfaces in the AFW system for all three Oconee units. We, therefore, consider the concern for a walkdown of nonseismically qualified areas to be resolved.

Based on the above, the staff concludes that the licensee has demonstrated adequate post-seismic event decay heat removal capability in accordance with the criteria of Generic Letter 81-14 by committing to correct identified deficiencies in the seismic qualification of the AFW system itself, and by demonstrating adequate seismically qualified alternative capability utilizing the SSF ASW pump and HPI pump (feed-and-bleed) in the event of loss of the AFW system as a result of seismically induced flooding. We, therefore, conclude that Oconee meets the requirements of GDC 2 and 34 for post-seismic shutdown decay heat removal capability and is, therefore, acceptable. A schedule for implementation of required modifications should be provided by January 5, 1987 with actions completed by January 1990. We consider MPA C-14, Seismic Qualification of the AFW System to be complete for Oconee, Units 1, 2 and 3.

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SALP INPUT
Seismic Qualification of the Auxiliary Feedwater System (MPA C-14)

Plant: Oconee Nuclear Station, Units 1, 2 and 3

1. Management Involvement and Control in Assuring Quality: Category 2

   Once made aware of the importance the staff placed on resolving this issue, Duke Power Company management expressed strong interest in gaining final resolution of the remaining concerns in the area of AFW system seismic qualification and provided the necessary control to assure quality responses. However, management control did not expedite response to the staff.

2. Approach to Resolution of Technical Issues from a Safety Standpoint: Category 2

   After additional discussions, the licensee finally recognized the significance of assuring post-seismic event decay heat removal capability, and demonstrated adequate knowledge for resolution of the remaining technical issues.

3. Responsiveness to NRC Initiatives: Category 3

   Even after management involvement, the licensee was not timely in providing the necessary additional information for resolution of identified staff concerns. Final resolution of MPA C-14 took over four years to accomplish.

4. Enforcement History: Not Applicable

5. Reporting and Analysis of Reportable Events: Not Applicable

6. Staff (Including management): Not Applicable

7. Training and Qualification Effectiveness: Not Applicable

Overall Rating: Category 2

# Proposed Rules

Federal Register
Vol. 51, No. 209
Wednesday, October 29, 1986



## NUCLEAR REGULATORY COMMISSION

## 10 CFR Part 50

## Leakage Rate Testing of Containments of Light-Water-Cooled Nuclear Power Plants

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Proposed rule.

**SUMMARY:** The Nuclear Regulatory Commission is proposing to amend its regulations to update the criteria and clarify questions of interpretation in regard to leakage rate testing of containments of light-water-cooled nuclear power plants. The purposed rule would aid the licensing and enforcement staff by eliminating conflicts, ambiguities, and lack of uniformity in the regulation of the inservice inspection program.

**DATE:** Comment period expires January 26, 1987. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except for comments received on or before this date.

**ADDRESSES:** Mail written comments to: U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch. Deliver comments to: Room 1121, 1717 H Street NW., Washington, DC, between 8:15 a.m. and 5:00 p.m. weekdays.

Copies of draft regulatory guide MS 021-5 may be obtained from the Nulear Regulatory Commission, Document Management Branch, Washington, DC 20555.

**FOR FURTHER INFORMATION CONTACT:** Mr. E. Gunter Arndt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 443-7893.

**SUPPLEMENTARY INFORMATION:**

### Background

Appendix J of 10 CFR Part 50 was originally issued for public comment as a proposed rule on August 27, 1971 (36 FR 17053); published in final form on February 14, 1973 (38 FR 4385); and became effective on March 16, 1973. The only amendment to this appendix since 1973 was a limited one, on Type B (penetration) test requirements that was published for comment on January 11, 1980 (45 FR 2330); published in final form September 22, 1980 (45 FR 62789); and became effective on October 22, 1980.

This revision of Appendix J has been in preparation for some time. It will provide greater flexibility in applying alternative requirements due to variations in plant design and reflects changes based on: (1) Experience in applying the existing requirements; (2) Advances in containment leak testing methods; (3) Interpretive questions; (4) Simplifying the text; (5) Various external/internal comments since 1973; and (6) Exemption requests received and approved.

The proposed revision is for the purpose of updating the existing regulation. Other related, longer term, and broader issues are currently under review by the NRC staff, such as containment function, degree of integrity required, and validation of that integrity under conditions other than postulated in this rule. In order to better understand its function and scope, assumptions inherent in Appendix J are presented as follow:

1. Certain levels of radiation exposure at the plant site boundary shall not be exceeded under (a) operating or (b) design basis accident conditions.
2. Certain levels of radiation exposure to plant operating personnel shall not be exceeded under (a) operating or (b) design basis accident conditions.
3. All four exposure levels (1a, 1b, 2a, 2b) may be different, but can be calculated.
4. Defense-in-depth will be used for protection against these levels of exposures. As the final barrier, a containment system is required in order to maintain any or all of these exposure limits.
5. The required degree of containment system leaktightness for design basis accidents can be (a) calculated, (b) specified, (c) built, (d) maintained, (e) inspected.
6. A generic inspection program can be defined that verifies the required leaktightness of the containment following construction and periodically throughout plant life.
7. NRC regulations should require such an inspection program, and define the test requirements and acceptance criteria.
8. A standard loss-of-coolant accident is assumed as the design basis accident. Since the containment isolation system is an engineered safety feature, only safety grade systems and components are relied upon to define the containment boundary that must be exposed to the containment pneumatic test pressure for the integrated leak rate test. In addition, all safety grade systems are assumed to be subject to a potential single active failure, and must be locally leak rate tested accordingly.
9. Pneumatic testing to peak calculated accident pressure is adequate without testing for, or at, accident temperatures or radiation levels.
10. Shielding tests need not be performed.
11. Periodic testing provides adequate confidence in the level of containment system integrity. Continuous monitoring of all individual isolation barriers is not necessary.

The scope of this revision to Appendix J is limited to corrections and clarifications, and excludes new criteria. However, this notice also addresses related, broader, longer term activities. Following is information of some of these other related activities that are not reflected in this proposed rulemaking.

In order to better identify the availability of containment leakage integrity, concepts of "continuous containment leakage monitoring" (such as negative containment operating pressure) and "relatively frequent gross containment integrity check" (such as a low pressure pumpup just prior to operation to check for openings) are under consideration by the NRC staff. These would identify large breaches of the containment system boundary, during or just prior to, normal operating conditions. It should be noted they would only test the normal operating containment atmosphere boundary, not the Appendix J, post-accident boundary including isolation valves. Comments on these or alternative concepts, and what effect, if any, they would have on the proposed Appendix J requirements, are also being solicited in the following section of this preamble.

Past practice has been to implement the provisions of Appendix J by means of licensees' technical specifications.

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Currently, a Technical Specification Improvement Project (TSIP) is underway to reevaluate the NRC's philosophy and utilization of the technical specifications. While the proposed revision described herein assumes implementation of Appendix J by licensee's technical specifications, the work of the TSIP may lead to some changes in this form of implementation.

Another program is presently being conducted to identify current NRC regulatory requirements that have marginal importance to safety and to recommend appropriate actions to modify or to eliminate these unnecessary requirements. A Federal Register notice was published on October 3, 1984, to announce the initiation of the program (49 FR 39066). As a part of the program, regulatory requirements associated with containment leaktightness are being evaluated. The risk and cost effectiveness of containment leaktightness requirements will be examined to determine their value with respect to plant safety and possible alternative requirements.

Any resulting changes to existing regulations will be made through normal rulemaking procedures, including ACRS review and public comment. Comments on the questions posed in this notice will also provide early, useful input to these associated activities.

## Invitation To Comment

Comments from all interested persons on all aspects of this revision and on the risk and cost effectiveness of containment leaktightness in general are requested by the comment expiration date in order that: (1) The final revision will reflect consideration of all points of view; and (2) The staff's assessment of the risk importance of containment leaktightness can benefit from such comments. Especially requested are comments which address the following questions:

(1) The extent to which there positions in the proposed rule are already in use;

(2) The extent to which those in use, and those not in use but proposed, are desirable;

(3) Whether there continues to be a further need for this regulation;

(4) Estimates of the costs and benefits of this proposed revision, as a whole and of its separate provisions;

(5) Whether present operating plants or plants under review should be given the opportunity to continue to meet the current Appendix J provisions if the proposed rule (reflecting consideration of public comments) becomes effective;

(6) If the existing rule or its proposed revision were completely voluntary, how many licensees would adopt either version in its entirety and why;

(7) Whether (a) all or part of the proposed Appendix J revisions would constitute a "backfit" under the definition of that term in the Commission's Backfit Rule, and (b) there are parts of the rule which do not constitute backfits, but which would aid the staff, licensees, or both;

(8) Since the NRC is planning a broader, more comprehensive review of containment functional and testing requirements in the next year or two, whether it is then still worthwhile to go forward with this proposed revision as an interim updating of the existing regulation;

(9) The advisability of referencing the testing standard (ANSI/ANS 56.8) in the regulatory guide (MS 021-5) instead of in the text of Appendix J;

(10) The value of collecting data from the "as found" condition of values and seals and the need for acceptance criteria for this condition;

(11) Whether the technical specification limits on allowable containment leakage should be relaxed and if so, to what extent and why, or if not, why not;

(12) What risk-important factors influence containment performance under severe accident conditions, to what degree these factors are considered in the current containment testing requirements, and what approaches should be considered in addressing factors not presently covered;

(13) What other approaches to validating containment integrity could be used that might provide detection of leakage paths as soon as they occur, whether they would result in any adjustments to the Appendix J test program and why;

(14) What effect "leak-before-break" assumption could have on the leakage rate test program. Current accident assumptions use instantaneous complete breaks in piping systems, resulting in a test program based on pneumatic testing of vented, drained lines. "Leak-before-break" assumptions presume that pipes will fail more gradually, leaking rather than instantly emptying.

(15) How to effectively adjust Type A test results to reflect individual Type B and C test results obtained from inspections, repairs, adjustments, or replacements of penetrations and valves in the years in between Type A tests. Such an additional criterion, currently outside the scope of this proposed revision, would provide a more meaningful tracking of overall containment leaktightness on a more continuous basis than once every several years. The only existing or proposed criterion for Type B and C tests performed outside the outage in which a Type A test is performed is that the sum of Type B and C tests must not exceed 60% of the allowable containment leakage. Currently being discussed by the NRC staff are:

a. All Type B and C tests performed during the same outage as a Type A test, or performed during a specified time period (nominally 12 months) prior to a Type A test, be factored into the determination of a Type A test "as found" condition.

b. If a particular penetration or valve fails two consective Type B or C tests, the frequency of testing that penetration must be increased until two satisfactory B or C tests are obtained at the nominal test frequency. Concurrently, existing requirements to increase the frequency of Type A tests due to consecutive "as found" failures are already being relaxed in the proposed revision of Appendix J. Instead, attention would be focused on correcting component degradation, no matter when tested, and the "as found" Type A test would reflect the actual condition of the overall containment boundary.

c. Increases or decreases in Type B or C "as found" test results (over the previous "as left" Type B or C test results) shall be added to or subtracted from the previous "as left" Type A test result.

If this sum exceeds 0.75 $L_a$ but is less than 1.0 $L_a$ measures shall be taken to reduce the sum to no more than 0.75 $L_a$. This will not be considered a reportable condition.

If this sum exceeds 1.0 $L_a$, measures shall be taken to reduce the sum to no more than 0.75 $L_a$. This will be considered a reportable condition.

The existing requirements that the sum of all Type B and C tests be no greater than 0.60 $L_a$ shall also remain in effect.

## Major Changes

The following are the major changes proposed in this rulemaking.

*1. Level of detail.* The level of detail addressed in the proposed revision of Appendix J has been limited. This revision of the regulation defines general containment system leakage test criteria.

*2. Editorial.* For increased clarity, an expanded and revised Table of Contents and set of definitions has been provided, conforming to current usage. The text has also been revised to conform to "plain English" objectives.

3. *Interpretations.* Some changes have been made to resolve past questions of interpretation (e.g., definitions of "containment isolation valves").

4. *Greater flexibility.* A major problem with Appendix J has been the lack of a provision for dealing with plants already built where design features are incompatible with Appendix J requirements (e.g., air lock testing). As a result, provision has been made in this revision for consideration by the NRC staff of alternative leakage test requirements when necessary.

5. *Type A test pressure.* The option of performing periodic reduced pressure testing in lieu of testing at full calculated accident pressure has been dropped. This change reflects the opinion that extrapolating low pressure leakage test results to full pressure leakage test results has turned out to be unsuccessful. Reasonable argument can be made for low pressure testing. However, the NRC staff believes that the peak calculated accident pressure (a) has always been the intended reference test pressure, (b) is consistent with the typical practice for NRC staff evaluations of accident pressure for the first 24 hours in accordance with Regulatory Guides 1.3 and 1.4, (c) provides at least a nominal check for gross low pressure leak paths that a low pressure leak does not provide for high pressure leak paths, (d) directly represents technical specification leakage rate limits, and (e) provides greater confidence in containment system leaktight integrity. For these reasons, the full, rather than reduced, pressure has been retained as the test pressure.

6. *Type A test frequency.* The test frequency has been uncoupled from the 10-year inservice inspection period used by the ASME Boiler & Pressure Vessel Code for mechanical systems. A different time base is used, but the frequency has remained essentially the same.

7. *Type A test duration.* The duration has been dropped from the test criteria in Appendix J. It is considered as part of the testing procedures, and is a function of the state of the testing technology and the level of confidence in it.

8. *Type A test "as is" clarification.* Appendix J originally noted in III.A.1(a) that the containment was to be ". . . tested in as close to the 'as is' condition as practical." This is re-emphasized and clarified by the explicit requirements that have been added to measure, record, and report "as found" and "as left" leakage rates.

9. *Type A test allowable leakage rate prorating.* Seventy-five percent of the allowable leakage rate represents the "as left" Type A test acceptance criterion, leaving 0.25 of the allowable leakage rate as a margin for deterioration until the time of the next regulatory scheduled Type A test, when the "as found" leakage rate criterion is 1.0 of the allowable leakage rate.

10. *Quantification of allowable leakage rates.* It should be noted that *no* change has been made to the way in which the allowable test leakage rates are quantified. The regulation still refers to the individual plant technical specifications for these values. Debate continues, however, on what these values should be and whether they can be generically specified, rather than individually specified for each site and plant.

11. *Refocusing of corrective actions.* When a reportable problem is identified, a Corrective Action Plan is to be submitted. It identifies the problem to the NRC staff, and notes the cause, what was or will be done to correct it, and what will be done to prevent its recurrence.

Increased local leakage testing frequency may be necessary. Appendix J originally addressed increased test frequency only for Type A tests. This revision applies adjustment of test frequency directly to identified problem areas.

12. *The final paragraph of the proposed amendment specifies a date by which an implementation schedule must be submitted, rather than by which it must be implemented.* This is because the ease with which licensees will be able to implement all the provisions of the amendment will be highly plant specific depending on plant design, outage and testing schedules, and amount of technical specification changes needed.

The separate views of Commissioner Frederic M. Bernthal follow:

The public should be aware of the fact that the Commission for over a year has attempted to adapt the Backfit Rule to *all* rulemaking, even rulemaking that has nothing to do with changes to powerplant hardware and the original intent of the Rule.

This rulemaking and the accompanying analysis illustrates the difficulty. When applied to human-factors rules, updating antiquated rules, and certain other rulemaking, the Backfit Rule continues to exact NRC resources wholly disproportionate to any conceivable benefit to the public. The record already shows cases where the Commission has been forced to sidestep a strict reading of the cost-benefit requirements and the ". . . substantial increase in overall protection . . ." threshold of the Backfit Rule, when it nevertheless finds broad agreement that a rulemaking is in the public interest (e.g. in the case of conversion of non-power reactors from HEU to LEU).

The public may therefore wish to comment directly on the question of whether the Commission should continue its attempts to apply the Backfit Rule to all rulemaking, or whether the Rule should be revoked as it applies to rulemaking activity per se.

Alternatively, the public may wish to consider whether the Commission should amend the Backfit Rule to waive the "substantial increase" provision, and to indicate explicitly that non-monetary benefits may be weighed by the Commission in the cost-benefit balance, when such considerations are found by the Commission to be in the public interest.

**Finding Of No Significant Environmental Impact: Availability**

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, if adopted, would not be a major Federal Action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required. There will be no radiological environmental impact offsite, but there may be an occupational radiation exposure onsite of about 3.0 man-rem per year of plant operation for inspection personnel (about 0.4% increase). Alternatives to issuing this revision were considered and found not acceptable. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 1717 H Street NW., Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from Mr. E. Gunter Arndt, Office of Nuclear Regulatory Research.

U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 443-7893.

**Paperwork Reduction Act Statement**

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

### Regulatory Analysis

The Commission has prepared a draft regulatory analysis on the proposed revision. The analysis examines the costs and benefits of the alternatives considered by the Commission. The draft analysis is available for inspection and copying in the NRC Public Document Room, 1717 H Street, NW., Washington, DC. The Commission requests public comment on the draft analysis. Comments may be submitted to the NRC as indicated under the ADDRESSES heading.

### Backfit Analysis

The Commission has prepared a backfit analysis on the proposed revision. The analysis is required under 10 CFR Part 50, § 50.109, as of October 21, 1985, for the management of backfitting for power reactors. The analysis is available for inspection and copying in the NRC Public Document Room, 1717 H Street NW., Washington, DC. The Commission requests public comment on the analysis. Comments may be submitted to the NRC as indicated under the ADDRESSES heading.

The analysis does not conclude that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit. It does conclude, however, that the direct and indirect costs of implementation are justified due to better, more uniform tests and test reports, greater confidence in the reliability of the test results, fewer exemption requests, and fewer interpretive debates. For these reasons, which are presented in greater detail in the backfit analysis, the Commission has decided to proceed with publication of the proposed rule for comment. The Commission's decision regarding promulgation of the rule, even though it may not provide a substantial increase in the overall protection of the public health and safety or the common defense and security, is tentative pending receipt of public comments on this issue.

### Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, (5 U.S.C. 605(b)), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

### List of Subjects in 10 CFR Part 50

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

### Related Regulatory Guide

The notice of availability of a draft regulatory guide on the same subject "Containment System Leakage Testing" (MS 021-5) is also being published elsewhere in this **Federal Register**. The draft regulatory guide contains specific guidance on acceptable leakage test methods, procedures, and analyses that may be used to implement these requirements and criteria.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

## PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

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

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 94-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273); 50.10 (a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); 50.10 (b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. Appendix J to Part 50 is revised to read as follows:

### Appendix J—Leakage Tests for Containments of Light-Water-Cooled Nuclear Power Plants



### I. Introduction

One of the conditions of all operating licenses for light-water-cooled power reactors as specified in § 50.54(o) of this part is that primary containments meet the leak test requirements set forth in this appendix. The tests ensure that (a) leakage through the primary containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the Technical Specifications and (b) inservice inspection of penetrations and isolation valves is performed so that proper maintenance and repairs are made during their service life. This appendix identifies the general requirements and acceptance criteria for preoperational and subsequent periodic leak testing.[1]

[1] Specific guidance concerning acceptable leakage test method, procedures, and analyses that may be used to implement these requirements and criteria will be provided in a regulatory guide that is being issued in draft form for public comment with the designation MS 021-5. Copies of the regulatory guide may be obtained from the Nuclear Regulatory Commission, Document Management Branch, Washington, DC 20555.

## II. Definitions

*Acceptance Criteria*

Standards against which test results are to be compared for establishing the functional acceptability of the containment system as a leakage limiting boundary.

*"As Found" Leakage Rate*

The leakage rate prior to any needed repairs or adjustments to the leakage barrier being tested.

*"As Left" Leakage Rate*

The leakage rate following any needed repairs or adjustments to the leakage barrier being tested.

*Containment Integrated Leak Rate Test (CILRT)*

The combination of a Type A test and its verification test.

*Containment Isolation System Functional Test*

A test to verify the proper performance of the isolation system by normal operation of the valves. For automatic containment isolation systems, a test of the automatic isolation system performed by actuation of the containment isolation signals.

*Containment Isolation Valve*

Any valve defined in General Design Criteria 55, 56, or 57 of Appendix A "General Design Criteria for Nuclear Power Plants," to this part.

*Containment Leak Test Program*

The comprehensive testing of the containment system that includes Type A, B, and C tests.

*Containment System*

The principal barrier, after the reactor coolant pressure boundary, to prevent the release of quantities of radioactive material that would have a significant radiological effect on the health of the public. It includes:

(1) The primary containment, including access openings and penetrations,

(2) Containment isolation valves, pipes, closed systems, and other components used to effect isolation of the containment atmosphere from the outside environs, and

(3) Those systems or portions of systems that by their functions extend the primary containment boundary to include their system boundary.

This definition does not include boiling water reactors' (BWR) reactor buildings or pressurized water reactors' (PWR) shield buildings. Also excluded from the provisions of this appendix are the interior barriers such as the BWR Mark II drywell floor and the drywell perimeters of the BWR Mark III and the PWR ice condenser.

*$L_a$ (weight percent/24 hr)*

The maximum allowable Type A test leakage rate in units of weight percent per 24-hour period at pressure $P_{ac}$ as specified in the Technical Specifications.

*$L_{am}$ (weight percent/24 hr)*

The measured Type A test leakage rate in units of weight percent per 24-hour period at pressure $P_{ac}$ obtained from testing the containment system in the state as close as practical to that that would exist under design basis accident conditions (e.g., vented, drained, flooded, or pressurized).

*Leak*

An opening that allows the passage of a fluid.

*Leakage*

The quantity of fluid escaping from a leak.

*Leakage Rate*

The rate at which the contained fluid escapes from the test volume at a specified test pressure.

*Maximum Pathway Leakage Rate*

The maximum leakage rate that can be attributed to a penetration leakage path (e.g., the larger, not total, leakage of two valves in series). This generally assumes a single active failure of the better of two leakage barriers in series when performing Type B or C tests.

*Minimum Pathway Leakage Rate*

The minimum leakage rate that can be attributed to a penetration leakage path (e.g., the smallest leakage of two valves in series). This is used when correcting the measured value of containment leakage rate from the Type A test ($L_{am}$) to obtain the overall integrated leakage rate and generally assumes no single active failure of redundant leakage barriers under these test conditions.

*Overall Integrated Leakage Rate*

The total leakage rate through all leakage paths, including containment welds, valves, fittings, and components that penetrate the containment system, expressed in units of weight percent of contained air mass at test pressure per 24 hours.

*$P_{ac}$ (psig)*

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the technical specifications.

*Periodic Leak Test*

Test conducted during plant operating lifetime.

*Preoperational Leak Test*

Test conducted upon completion of construction of a primary or secondary containment, including installation of mechanical, fluid, electrical, and instrumentation systems penetrating these containment systems, and prior to the time containment integrity is required by the Technical Specifications.

*Primary Containment*

The structure or vessel that encloses the major components of the reactor coolant pressure boundary as defined in § 50.2(v) of this part and is designed to contain accident pressure and serve as a leakage barrier against the uncontrolled release of radioactivity to the environment. The term "containment" as used in this appendix refers to the primary containment structure and associated leakage barriers.

*Structural Integrity Test*

A pneumatic test that demonstrates the capability of a primary containment to withstand a specified internal design pressure load.

*Type A Test*

A test to measure the containment system overall integrated leakage rate under conditions representing design basis loss-of-coolant accident containment pressure and systems alignments (1) after the containment system has been completed and is ready for operation and (2) at periodic intervals thereafter. The verification test is not part of this definition—see CILRT.

*Type B Test*

A pneumatic test to detect and measure local leakage through the following containment penetrations:

(1) Those whose design incorporates resilient seals, gaskets, sealant compounds, expansion bellows, or fitted with flexible metal seal assemblies.

(2) Air locks, including door seals and door operating mechanism penetrations that are part of the containment pressure boundary.

*Type C Test*

A pneumatic test to measure containment isolation valve leakage rates. *Verification Test*

Test to confirm the capability of the Type A test method and equipment to measure $L_{am}$.

## III. General Leak Test Requirements

*A. Type A Test*

(1) Preoperational Test. A preoperational Type A test must be conducted on the containment system and must be preceded by:

(a) Type B and Type C tests.

(b) A structural integrity test.

(2) Periodic Test. A periodic Type A test must be performed on the containment system.

(3) Test Frequency. Unless a longer interval is specifically approved by the NRC staff, the interval between the preoperational and first periodic Type A tests must not exceed three years, and the interval between subsequent periodic Type A tests must not exceed four years. If the initial fuel loading is delayed so that the three-year interval between the first preoperational test and the first periodic test is exceeded, another preoperational test will be necessary. If such an additional preoperational Type A test or an additional Type A test required by Section III.A.8 or IV.A. of this appendix is performed, the Type A test interval may be restarted.

(4) Test Pressure. The Type A test pressure must be equal to or greater than $P_{ac}$ at the start of the test but must not exceed the containment design pressure and must not fall more than 1 psi below $P_{ac}$ for the duration of the test, not including the verification test. The test pressure must be established relative to the external pressure of the containment. This may be either atmospheric pressure or the subatmospheric pressure of a secondary containment.

(5) Pretest Requirements. Closure of containment isolation valves for the Type A

test must be accomplished by normal operation and without any preliminary exercising or adjustments for the purpose of improving performance (e.g., no tightening of valve after closure by valve motor). Repairs of malfunctioning or leaking valves must be made as necessary. Information on valve leakage that requires corrective action prior to, during, or after the test (see Section V.B.) must be included in the report submitted to the Commission as specified in Section VI of this appendix.

(6) Verification Test. A leakage rate verification test must be performed after a Type A test in which the leakage rate meets the criterion in III.A.(7)(b)(ii). The verification test selected must be conducted for a duration sufficient to establish accurately the change in leakage rate between the Type A and verification tests. The results of the Type A test are acceptable if the sum of the verification test imposed leadkage and the containment leakage rate calculated from the Type A test ($L_{am}$) does not differ from the leakage rate calculated from the verification test by more than $\pm 0.25\,L_a$.

(7) Acceptance Criteria.

(a) For the preoperational Type A Test, the "as left" leakage rate must not exceed $0.75L_a$, as determined by a properly justified statistical analysis. The "as found" leakage rate does not apply to the preoperational test.

(b) For each periodic Type A test, the leakage rate, as determined by a properly justified statistical analysis, must not exceed:

(i) $L_a$, for the "as found" condition.

(ii) $0.75L_a$, for the "as left" condition.

(c) In meeting these Type A test acceptance criteria, isolation, repair, or adjustment to a leakage barrier that may affect the leakage rate through that barrier is permitted prior to or during the Type A test provided:

(i) All potential leakage paths of the isolated, repaired, or adjusted leakage barrier are locally leak testable, and

(ii) the local leakage rates are measured before and after the isolation, repair, or adjustment and are reported under Section VI of this appendix.

(iii) All changes in leakage rates resulting from isolation, repair, or adjustment of leakage barriers subject to Type B or Type C testing are determined using the minimum pathway leakage method and added to the Type A test result to obtain the "as found" and "as left" containment leakage rates.

(d) The effects of isolation, repair, or adjustments to the containment boundary made after the start of the Type A test sequence on the Type A test results must be quantified and the appropriate analytical corrections made (this includes tightening valve stem packing, additional tightening of manual valves, or any action taken that will affect the leakage rates).

(8) Retesting.

(a) If, for any periodic Type A test, the as found leakage rate fails to meet the acceptance criterion of $1.0L_a$, a Corrective Action Plan that focuses attention on the cause of the problem must be developed and implemented by the licensee and then submitted together with the Containment Leak Test Report as required by Section VI of this appendix. The test schedule applicable to subsequent Type A tests (III.A.(3)) shall be submitted to the NRC staff for review and approval. An as left Type A test that meets the acceptance criterion of $0.75L_a$ is required prior to plant startup.

(b) If two consecutive periodic as found Type A tests exceed the as found acceptance criterion of $1.0L_a$:

(i) Regardless of the periodic retest schedule of III.A.(3), a Type A test must be performed at least every 24 months (based on the refueling cycle normally being about 18 months) unless an alternative leakage test program is acceptable to the NRC staff on some other defined basis. This testing must be performed until two consecutive periodic "as found" Type A tests meet the acceptance criterion of $1.0L_a$ after which the retest schedule specified in III.A.(3) may be resumed.

(ii) Investigation as to the cause and nature of the Type A test failure might indicate that an alternative leakage test program such as more frequent Type B or Type C testing may be more appropriate than the performance of two consecutive successful Type A leakage tests. The licensee may then submit a Corrective Action Plan and an alternative leakage test program proposal for NRC staff review. If this submittal is approved by the NRC staff, the licensee may implement the corrective action and alternative leakage test program in lieu of one or both of the Type A leakage tests required by Section III.A.(8)(b)(i).

(9) Permissible periods for testing. The performance of Type A tests must be limited to periods when the plant facility is secured in the shutdown condition under the administrative controls and safety procedures defined in the license.

B *Type B Test*

(1) Frequency.

(a) Type B tests, except tests for air locks, must be performed on containment penetrations during shutdown for refueling or at other convenient intervals but in no case at intervals greater than 2 years. If opened following a Type A or B test, containment penetrations subject to Type B testing must be Type B tested prior to returning the reactor to an operating mode requiring containment integrity.

(b) For containment penetrations employing a continuous leakage monitoring system that is at a pressure not less than $P_{ac}$, leakage readings of sufficient sensitivity to permit comparison with Type B test leak rates must be taken at intervals specified in the Technical Specifications. These leakage readings must be part of the Type B reporting of VI.A. When practical, continuous leakage monitoring systems must not be operating or pressurized during Type A tests. If the continuous leakage monitoring system cannot be isolated, such as inflatable air lock door seals, leakage into the containment must be accounted for and the Type A test results corrected accordingly.

(2) Pressure. Type B tests must be conducted, whether individually or in groups, at a pneumatic pressure not less than $P_{ac}$ except as provided in paragraph III.B.(3)(b) of this section or in the Technical Specifications.

(3) Air Locks.

(a) Initial and periodic tests. Air locks must be tested prior to initial fuel loading and at least once each 6-month interval thereafter at an internal pressure not less than $P_{ac}$. Alternatively, if there have been no air lock openings within 6 months of the last successful test at $P_{ac}$, this interval may be extended to the next refueling outage or airlock opening (but in no case may the interval exceed 2 years). Reduced pressure tests must continue to be performed on the air lock or its door seals at 6-month intervals. Opening of the air lock for the purpose of removing air lock testing equipment following an air lock test does not require further testing of the air lock.

(b) Intermediate tests must be conducted as follows:

(i) Air locks opened during periods when containment integrity is required by the plant's Technical Specifications must be tested within 3 days after being opened. For air lock doors opened more frequently than once every 3 days, the air lock must be tested at least once every 3 days during the period of frequent openings. Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications need not be repeatedly tested during such periods. However, they must be tested prior to the plant requiring containment integrity. For air lock doors having testable seals, testing the seals fulfills the intermediate test requirements of this paragraph. In the event that this intermediate testing cannot be done at $P_{ac}$, the test pressure must be as stated in the Technical Specifications.

(ii) Whenever maintenance other than on door seals has been performed on an air lock, a complete air lock test at a test pressure of not less than $P_{ac}$ is required, if that maintenance involved the pressure retaining boundary.

(iii) Air lock door seal testing or reduced-pressure testing may not be substituted for the initial or periodic full-pressure test of the entire air lock required in paragraph III.B.(3)(a) of this Section.

(4) Acceptance Criteria.

(a) The sum of the as found or as left Type B and C test results must not exceed $0.60L_a$ using maximum pathway leakage and including leakage rate readings from continuous leakage monitoring systems.

(b) Leakage measurements are acceptable if obtained through component leakage surveillance systems (e.g., continuous pressurization of individual or clustered containment components) that maintain a pressure not less than $P_{ac}$ at individual test chambers of those same containment penetration during normal reactor operation. Similar penetrations not included in the component leakage surveillance system are still subject to individual Type B tests.

(c) An air lock, penetration, or set of penetrations that fails to pass a Type B test must be retested following determination of cause and completion of corrective action. Corrective action to correct the leak and to prevent its future recurrence must be developed and implemented.

(d) Individual acceptance criteria for all air lock tests must be stated in the Technical Specifications.

*C. Type C Test*

(1) Frequency. Type C tests must be performed on containment isolation valves during each reactor shutdown for refueling or at other convenient intervals but in no case at intervals greater than 2 years.

(2) Pressure/Medium.

(a) Containment isolation valves unless pressurized with a qualified water seal system must be pressurized with air or nitrogen at a pressure not less than $P_{ac}$.

(b) Containment isolation valves, that are sealed with water from a qualified seal system, must be tested with water at a pressure not less than 1.10 $P_{ac}$.

(3) Acceptance Criteria.

(a) The sum of the as found or as left Type B and C test results must not exceed $0.60L_a$ using maximum pathway leakage and including leakage rate readings from continuous leakage monitoring systems.

(b) Leakage from containment isolation valves that are sealed with water from a seal system may be excluded when determining the combined Type B and C leakage rate if:

(i) The valves have been demonstrated to have leakage rates that do not exceed those specified in the Technical Specifications, and

(ii) The installed isolation valve seal system inventory is sufficient to ensure the sealing function for at least 30 days at a pressure of 1.10 $P_{ac}$.

(4) Valves That Need Not Be Type C Tested.

(a) A containment isolation valve need not be Type C tested if it can be shown that the valve does not constitute a potential containment atmosphere leak path during or following an accident, considering a single active failure of a system component.

(b) Other valves may be excluded from Type C testing only when approved by the NRC staff under the provisions of paragraph VII.A.

**IV. Special Leak Test Requirements**

*A. Containment Modification or Maintenance*

Any modification, repair, or replacement of a component that is part of the containment system boundary and that may affect containment integrity must be followed by either a Type A, Type B, or Type C test. Any modification, repair, or replacement of a component subject to Type B or Type C testing must also be preceded by a Type B or Type C test. The measured leakage from this test must be included in the report to the Commission required by Section VI of this appendix. Following structural changes or repairs that affect the pressure boundary, the licensee shall demonstrate whether or not a structural integrity test is needed prior to the next Type A test. The acceptance criteria of paragraph III.A.(7), III.B.(4), or III.C.(3) of this appendix, as appropriate, must be met. Type A testing of certain minor modifications, repairs, or replacements may be deferred to the next regularly scheduled Type A test if local leakage testing is not possible and visual (leakage) examinations or nondestructive examinations have been conducted. These shall include: Welds of attachments to the surface of the steel pressure retaining boundary; Repair cavities the depth of which does not penetrate the required design steel wall by more than 10%; Welds attaching to the steel pressure retaining boundary penetrations the nominal diameter of which does not exceed one inch.

*B. Multiple Leakage Barrier or Subatmospheric Containments*

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

**V. Test Methods, Procedures, and Analyses**

*A. Type A, B, and C Test Details*

Leak test methods, procedures, and analyses for a steel, concrete, or combination steel and concrete containment and its penetrations and isolation valves for light-water-cooled power reactors must be referenced or defined in the Technical Specifications.

*B. Combination of Periodic Type A, B, and C Tests*

Type B and C tests are considered to be conducted in conjunction with the periodic Type A test when performed during the same outage as the Type A test. The licensee shall perform, record, interpret, and report the tests in such a manner that the containment system leak-tight status is determined on both an as found basis and an as left basis, i.e., its leak status prior to this periodic Type A test together with the related Type B and C tests and its status following the conclusion of these tests.

**VI. Reports**

*A. Submittal*

1. The preoperational and periodic Type A tests, including summaries of the results of Type B and C tests conducted in conjunction with the Type A test, must be reported in a summary technical report sent not later than 3 months after the conduct of each test to the Commission in the manner specified in § 50.4. The report is to be titled "Containment Leakage Test."

2. Reports of periodic Type B and C tests conducted at intervals intermediate to the Type A tests must also be submitted to the NRC in the manner specified in § 50.4 and at the time of the next Type A test submittal. Reports must be submitted to the NRC Regional Administrator within 30 days of completion of any Type B or C tests that fail to meet their as found acceptance criteria.

*B. Content*

A Type A test Corrective Action Plan, when required under paragraph III.A.(8) of this appendix, must be included in the report. Any corrective action required for those Type B and C tests included as a part of the Type A test sequence must also be included in the reqort.

**VII. Application**

*A. Applicability*

The requirements of this appendix apply to all operating nuclear power reactor licensees as specified in § 50.54(o) of this part unless it can be demonstrated that alternative leak test requirements (e.g., for certain containment designs, leakage mitigation systems, or different test pressures not specifically addressed in this appendix) are demonstrated to be adequate on some other defined basis. Alternative leak test requirements and the basis for them will be made a part of the plant Technical Specifications if approved by the NRC staff.

*B. Effective Date*

This appendix is effective (30 days after publication of the final rule). By (insert a date 180 days after the effective date of this revision), each licensee and each applicant for an operating license shall submit a plan to the Director of the Office of Nuclear Reactor Regulation for implementing this appendix. This submittal must include an implementation schedule, with a final implementation no later than (insert a date 48 months after the effective date of this revision). Until the licensee finally implements the provisions of this revision, the licensee shall continue to use in their entirety the existing Technical Specifications and the Appendix J on which they are based. Thereafter, the licensee shall use in their entirety this revision and the Technical Specifications conforming to this revision.

Dated at Washington, DC, this 22d day of October, 1986.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

*Secretary of the Commission.*

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**DEPARTMENT OF TRANSPORTATION**

**Federal Aviation Administration**

**14 CFR Part 39**

**[Docket No. 86-CE-49-AD]**

**Airworthiness Directives; Beech 99 and 100 Series Airplanes**

**AGENCY:** Federal Aviation Administration (FAA), DOT.

**ACTION:** Notice of Proposed Rulemaking (NPRM).

**SUMMARY:** This Notice proposes to adopt a new Airworthiness Directive (AD), applicable to Beech 99 and 100 Series airplanes. The proposal would require inspection and replacement of rivets which attach each elevator outboard hinge to the stabilizer. Loose