PROPOSED RULE PR-50
Gen. Design Criteria

# ATOMICS INTERNATIONAL

*A Division of North American Rockwell Corporation*

September 25, 1967

DOCKETED
USAEC
SEP 29 1967
Office of the Secretary
Public Proceedings
Branch

*In reply refer—*

67AT-5374

Secretary
U. S. Atomic Energy Commission
Washington, D.C. 20545

Gentlemen:

The revised set of proposed General Design Criteria, which were published in the Federal Register on July 11, 1967, for public comment, represents the results of a great deal of very fruitful effort to develop standards to assist in the preparation of applications for nuclear power plant construction permits. The early release of the first set of criteria developed by the regulatory staff, with the request for comments, initiated the extensive efforts recognized as necessary for effective evolution and development of the criteria. These resulting criteria, which reflect the public comments and suggestions, represent a significant improvement, both in organization and format and in content, over the initial criteria published in 1965. They offer considerably more and better guidance for the preparation of applications for nuclear power plant construction permits and operating licenses.

Our review has resulted in a number of comments and recommendations which are outlined below. Our more general comments are followed by those specifically directed to the individual criteria by number.

First we recommend that in adoption of the proposed criteria as a part of 10 CFR 50, they be more specifically directed to and required of large pressurized and boiling water reactors. This approach in the application would reduce the possibility of ritualistic adherence by reviewers to the requirements of the criteria when considering reactor types other than those for which the criteria were specifically developed. Detailed implementation of the criteria for other reactor types, and particularly for the advanced reactors now receiving major attention, can then proceed in whatever manner is most appropriate for the reactor without preconceived conclusions from the results of application to the water reactors. Also this more specific application to water reactors will reduce the possibility of their misuse by intervenors in public hearings for other reactor types.

P. O. Box 309, Canoga Park, California 91304
Cable Address: ATOMICS

The proposed criteria appear to be extremely qualitative in a number of areas. For example, we note the use of words and phrases such as: "impairing of safety" (Criterion 4), "acceptable fuel damage limits" Criteria 6 and 14), "appropriate margins" (Criterion 6), "exceedingly low probability" (Criterion 9), "high functional reliability" (Criteria 19 and 38), "sufficient" (Criterion 20), "necessary" (Criterion 20), "considerable margin" (Criterion 32), "limited allowances" (Criterion 33), "abundant" and "negligible" (Criterion 44), "considerable margin" (Criterion 49), "as close to design as practicable" (Criteria 61 and 65), "reliable" (Criterion 67), "undue amounts" (Criterion 69), and "high population density for very large cities" (Criterion 70). While we recognize that development of effective definitions of these types of terms is a very difficult task, we wish to encourage a strong continuing effort to define the terms quantitatively and then to include a section on definitions as an integral part of the criteria.

Our specific comments on the individual criteria are identified below by each criterion number.

2. Some quite specific criteria have been developed and applied to such natural phenomena as tornadoes and earthquakes in previous reactor application reviews. Including examples of this kind of guidance would be helpful to applicants. We also recommend that, in addition to the two items cited, the design bases established as a result of this criterion reflect the results of analyses which include not only the quantitative severity of the natural phenomena but also their probability of occurrence.

4. The implication that any degradation or impairment of safety is unacceptable and should be removed.

5. It might be noted that the records should be accessible subsequent to the occurrence of an accident.

8. We believe that it is unnecessary to require the overall power coefficient to be not positive in the power operating range. It is quite possible for the overall coefficient to be positive, and there be no unacceptable safety problem. For example, in a sodium graphite reactor, the coefficient has a prompt negative component together with a positive component with a long time constant. This results in an overall positive coefficient, but the negative portion of the coefficient is large enough and fast enough to assure

satisfactory control and safety. In fact, the lack of an overall negative coefficient is an advantage, since compensation for a large temperature and power defect in the reactivity is not required.

10. It is entirely conceivable that containment, as used today for water reactors, may not be required for other types of reactors currently under development. It would seem appropriate to give some recognition now to this in this criterion.

11. The basic requirement here is the provision of a control room that will remain habitable and will provide capability to shut the reactor down and maintain it in a safe condition. Application of the radiation exposure limits in 10 CFR 20 in this criterion is unduly stringent and is unnecessary. The 10 CFR 20 limits are for normal operations and should not be required in "accident conditions."

13. The requirement for monitoring the fission process for "... all conditions that can ... cause variations in reactivity" is too inclusive in this context. The examples given are simple and of external origin. More subtle conditions could be, e.g., fuel motion during life, changes in core geometry, etc. It may not be possible to monitor these conditions directly. What is important is monitoring of reactivity, and a predictive analysis by means of which observations and predictions can be compared, and any anomalies identified.

14. We submit that it is unnecessary for all core protection systems "to act automatically."

16. This criterion should require monitoring for leakage of reactor coolant; monitoring the "reactor coolant pressure boundary" is unnecessary.

20. The bases for determining when two different operating principles are necessary should be included here.

28. It is not necessary for two reactivity control systems to act fast enough to prevent exceeding acceptable fuel damage. Hence, we recommend deletion of "... including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits."

...

29. Shutdown margins greater than the worth of the most effective control rod appear inconsistent with the fact that reactors now being licensed have in excess of 100 such rods. We suggest the criterion be directed to providing shutdown margins greater than the maximum worth of any one gang of rods which can be driven or controlled by an operator or the control system.

36. We would point out that, except for financial risk, the requirements of this criterion are unnecessary if failure of the coolant boundary does not result in loss of coolant and subsequent core failure. Hence, application of this to low pressure coolant systems can be relaxed significantly.

39. Requirements for offsite power should be deleted, since adequate onsite power systems must always be required for emergency operation of the engineered safety features.

42. Here, it should be recognized that the loss-of-coolant accidents may not be design basis accidents for other power reactors for which these criteria are generally applicable.

44. We believe that the extent of independence and redundancy outlined here for the emergency core cooling systems is not necessary for low pressure systems. Also we question the necessity for "preferably of different design principles."

66. The second sentence should be replaced with "Inherent means should be used where practicable."

67. The criterion should be revised to require the design to be based on preventing exposures in excess of 10 CFR 20 limits.

69. The criterion should require that containment be provided if radioactivity releases due to accidents lead to public exposure in excess of 10 CFR 20 limits.

We believe your consideration of our comments will lead to further improvements in the General Design Criteria. If there are questions, or if we can provide further clarification, we shall be pleased to do so.

Sincerely yours,

J. J. Flaherty
President
Atomics International Division

10/8/97

# POLICY ISSUE
## (Information)

July 21, 1997 SECY-97-155

FOR: The Commissioners

FROM: L. Joseph Callan
Executive Director for Operations

SUBJECT: STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

PURPOSE:

To inform the Commission of the action staff is taking associated with exemptions from 10 CFR 70.24 for commercial nuclear power plants.

BACKGROUND:

The Commission's regulations in 10 CFR 70.24 require that each licensee authorized to possess more than a small amount of special nuclear material (SNM) maintain in each area in which such material is handled, used, or stored a criticality monitoring system "using gamma- or neutron-sensitive radiation detectors [minimum of two detectors] which will energize clearly audible alarm signals if accidental criticality occurs." The regulation also specifies sensitivity requirements for these monitors and details the training that licensees must conduct in connection with criticality monitor alarms. The purpose of 10 CFR 70.24 is to ensure that, if a criticality were to occur during the handling of SNM, personnel would be alerted to that fact and would take appropriate action.

Most nuclear power plant licensees were granted exemptions from 10 CFR 70.24 during the construction of their plants as part of the Part 70 license issued to permit the receipt of the initial core. Generally, these exemptions were not explicitly renewed when the Part 50 operating license, which now contained the combined Part 50 and Part 70 authority, was issued.

CONTACT:
S. Singh Bajwa, NRR
(301) 415-1013

In August 1981. the Tennessee Valley Authority (TVA). in the course of reviewing the operating licenses for its Browns Ferry facilities. noted that the exemption to 10 CFR 70.24 that had been granted during the construction phase had not been explicitly granted in the operating license. By letters dated August 11. 1981. and August 31. 1987. TVA requested an exemption from 10 CFR 70.24. On May 11. 1988. NRC informed TVA that "the previously issued exemptions are still in effect even though the specific provisions of the Part 70 licenses were not incorporated into the Part 50 license." Recently. the Office of the General Counsel (OGC) determined that. in cases where a licensee received the exemption as part of the Part 70 license issued during the construction phase. both the Part 70 and Part 50 licenses should be examined to determine the status of the exemption. OGC's view is that. where the licenses and SERs are silent. an exemption expires with the expiration of the Part 70 license.

As part of NRC's effort to achieve regulatory improvement in granting exemptions to regulations (SECY-96-147. dated July 1. 1996). the Office of Nuclear Reactor Regulation (NRR) conducted a survey. in the summer of 1996. to determine licensee compliance with 10 CFR 70.24. Project managers reviewed the licenses for their assigned plants and contacted the licensees to determine if licensees satisfied the requirements of 10 CFR 70.24 or if an exemption to the requirements of the regulation had been granted. This survey indicated that 38 plants are in compliance. 6 plants are working on coming into compliance. 29 plants have exemptions. and 37 plants do not yet have exemptions; of these 37. 30 had already submitted exemption requests for staff review. Of the 29 plants that have exemptions. 13 received their exemption upon issuance of their operating license and 16 as the result of an exemption request.

DISCUSSION

At a commercial nuclear power plant. there are only three locations where amounts of SNM sufficient to cause a criticality may be found: the reactor core. the fresh fuel storage area. and the spent fuel pool. SNM other than fuel. such as in fission chamber neutron detectors and in neutron sources. may also be found in some laboratory and storage locations of these plants. but an inadvertent criticality is not considered credible in these areas due to the amount and configuration of the SNM. The SNM that could be assembled into a critical mass at a commercial nuclear power plant is in the form of nuclear fuel. This fuel is not enriched beyond 5.0 weight percent (wt %) uranium-235 (U-235) and commercial nuclear plant licensees have procedures and design features that prevent inadvertent criticality. The inadvertent criticality with which 10 CFR 70.24 is concerned could only occur during fuel-handling operations.

The staff concludes that, when one considers the administrative and design controls established and maintained at nuclear power plants, the criticality monitoring requirement of 10 CFR 70.24 is not necessary. At power reactor facilities with uranium fuel enriched to no greater than 5 wt% U-235, the SNM in the fuel assemblies cannot go critical without both a critical configuration and the presence of a moderator. The SNM in the reactor core is intended to become critical and need not be subjected to the provisions of 10 CFR 70.24. The fresh fuel storage array and the spent fuel pool are designed and analyzed to prevent inadvertent criticality, even in the presence of an optimal density of unborated moderator. Inadvertent criticality during fuel handling is precluded by limitations on the number of fuel assemblies permitted out of storage at the same time. In addition, General Design Criterion (GDC) 62 in Appendix A to 10 CFR Part 50 reinforces prevention of criticality in fuel storage and handling. Fuel handling at power reactor facilities occurs only under strict procedural control and supervision, including the use of certified fuel handlers.

In contrast, at fuel fabrication facilities, any number of these fuel assemblies may be moved many times each day. Fuel fabrication facilities handle SNM in various configurations. Although handling of this material is controlled by procedures, the variety of forms of SNM and the frequency with which it is handled increase the possibility of an inadvertent criticality.

In conclusion, fuel fabrication facilities are significantly different from reactor facilities in their handling of SNM. The staff considers a fuel-handling accidental criticality at a commercial nuclear power plant to be extremely unlikely due to administrative and design controls. Therefore, imposition of the 10 CFR 70.24 criticality monitoring requirement on licensees of operating reactors is not necessary as long as design and administrative controls are maintained.

PLANNED ACTION:

In accordance with the analysis set forth above, the staff has developed seven criteria (attached) to review exemption requests. Generally, these are the criteria that have been used in the past in processing exemptions to 10 CFR 70.24. The staff believes that most of the plants meet these criteria. However, if a facility does not meet any one of the criteria, the staff will request the licensee to justify deviations from any criteria that cannot be met.

In an effort to gain efficiencies and to expedite staff review of the 30 in-house exemption requests, the staff is planning to ask licensees to voluntarily supplement their exemption requests with a response that verifies that their facility meets the criteria. A similar approach will be taken with the licensees of seven operating plants that do not meet the requirements of 10 CFR 70.24 and have not yet requested exemption from the requirements of that section.

Licensees seeking an exemption from the requirements of 10 CFR 70.24 but have not yet submitted exemption requests will be asked to submit these requests within 60 days, together with statements either confirming their compliance with the staff's criteria or explaining their justification for any deviations.

RULEMAKING ACTION:

NRR and NMSS staff, with assistance from the Office of Nuclear Regulatory Research, are developing a rulemaking plan for modifying 10 CFR 70.24 to address the long-term issue of recurring exemptions to the regulation. This plan will be submitted to the Commission under separate cover by the end of August 1997.

ENFORCEMENT HISTORY:

The staff has issued approximately 20 Notices of Violation for failures by reactor licensees to meet the provisions of 10 CFR 70.24. These violations have been categorized as Severity Level IV violations and Non-Cited Violations. The staff has reconsidered these actions and concluded in light of the issues discussed in this paper that, although violations did occur, it is appropriate to exercise enforcement discretion generally in keeping with Section VII B.6 of the Enforcement Policy. The bases for exercising this discretion are the lack of safety significance of the failure to meet 10 CFR 70.24, provided controls are in place to ensure compliance with GDC 62; the failure of the NRC staff to recognize the need for an exemption during the licensing process; the NRC public position on this matter, as reflected in its letter of May 11, 1988, to TVA concerning the lack of a need for an exemption at Browns Ferry; and the position underlying the staff's intent to embark on rulemaking to amend 10 CFR 70.24 to provide for administrative controls in lieu of criticality monitors. Therefore, the staff intends to withdraw the issued violations. The staff does not intend to take further enforcement action for failure to meet 10 CFR 70.24 provided the licensee obtains an exemption to this regulation before the next receipt of fresh fuel or before the next planned movement of fresh fuel.

CONCLUSION:

The staff intends to take the actions noted ten days from the date of this paper.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection.

RESOURCES:

The paper provides the staff's action related to 70.24 exemptions and does not involve changes in resource requirements.

INFORMATION TECHNOLOGY:

No new anticipated impacts on information management or information technology are anticipated as a result of implementing the actions discussed in this paper.

GENERIC REQUIREMENTS:

The Committee to Review Generic Requirements was not requested to review this paper, since the planned staff action does not introduce a generic requirement.

L. Joseph Callan
Executive Director
for Operations

Attachment: Criteria for evaluating 70.24 Exemption Requests

DISTRIBUTION:
Commissioners
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EDO
REGIONS
SECY

CRITERIA FOR EVALUATING 70.24 EXEMPTION REQUESTS

1. Plant procedures do not permit more than one pressurized-water reactor or three boiling-water reactor fuel assemblies to be out of an approved storage configuration at one time.

2. The k-effective of the fresh fuel storage racks filled with fuel of the maximum permissible U-235 enrichment and flooded with pure water does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

3. If optimum moderation of fuel in the fresh fuel storage racks occurs when the fresh fuel storage racks are filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation does not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

4. The k-effective of spent fuel storage racks filled with fuel of the maximum permissible U-235 enrichment and flooded with pure water does not exceed 0.95, at a 95-percent probability, 95-percent confidence level.

5. The quantity of SNM other than nuclear fuel stored on site in any given area is less than the quantity necessary for a critical mass.

6. Radiation monitors, as required by GDC 63, are provided in fuel storage and handling areas to detect excessive radiation levels and to initiate appropriate safety actions.

7. The maximum nominal U-235 enrichment is 5 wt%.

ATTACHMENT

OFFICE OF THE SECRETARY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555-0001

August 19, 1997

10/8/97 [illegible]

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John C. Hoyle, Secretary

SUBJECT: STAFF REQUIREMENTS - SECY-97-155 - STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

The Commission has approved the staff's proposed actions for processing 10 CFR 70.24 exemptions and for addressing facilities which do not meet the requirements of 10 CFR 70.24 but which have not yet requested an exemption. The Commission has approved the staff's proposed actions for processing previous enforcement actions.

The Commission has also approved rulemaking to alleviate the 10 CFR 70.24 exemptión problem. The staff should proceed with direct final rulemaking on this matter and inform the Commission of the milestones and completion dates for this rulemaking.
(EDO) (SECY Suspense: 9/26/97)

The staff should issue appropriate generic communications regarding compliance and enforcement for the interim period until the rule can be appropriately corrected.
(EDO) (SECY Suspense: 10/17/97)

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
OGC
CIO
CFO
OCA
OIG

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICE OF THE
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cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
OGC
CIO
CFO
OCA
OIG

SECY NOTE: THIS SRM AND SECY-97-155 ARE ENFORCEMENT RELATED AND WILL BE LIMITED TO NRC UNLESS THE COMMISSION DETERMINES OTHERWISE.

OFFICE OF THE SECRETARY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 19, 1997

RELEASED TO THE PDR
10/8/97 date
initials

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cc: Chairman Jackson
Commissioner Dicus
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OIG

OFFICE OF THE SECRETARY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001



August 19, 1997

# COMMISSION VOTING RECORD

**DECISION ITEM:** SECY-97-155

**TITLE:** STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

The Commission (with all Commissioners agreeing) approved the subject paper as recorded in the Staff Requirements Memorandum (SRM) of August 19, 1997.

This Record contains a summary of voting on this matter together with the individual vote sheets, views and comments of the Commissioners, and the SRM of August 19, 1997.

John C. Hoyle
Secretary of the Commission

Attachments:
1. Voting Summary
2. Commissioner Vote Sheets
3. Final SRM

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
OGC
EDO

# VOTING SUMMARY - SECY-97-155

## RECORDED VOTES

| | APRVD | DISAPRVD | ABSTAIN | NOT PARTICIP | COMMENTS | DATE |
|---|---|---|---|---|---|---|
| CHRM. JACKSON | | X | | | X | 8/8/97 |
| COMR. DICUS | X | | | | X | 8/13/97 |
| COMR. DIAZ | X | | | | X | 8/5/97 |
| COMR. McGAFFIGAN | X | | | | X | 8/12/97 |

## COMMENT RESOLUTION

In their vote sheets, all Commissioners approved a portion of the staff's proposed actions but preferred an expedited rulemaking and agreed on the use of a direct final rulemaking. Subsequently, the comments of the Commission were incorporated into the guidance to staff as reflected in the SRM issued on August 19, 1997.

# NOTATION VOTE

## RESPONSE SHEET

**TO:** John C. Hoyle, Secretary

**FROM:** CHAIRMAN JACKSON

**SUBJECT:** SECY-97-155 - STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

Approved ____ Disapproved x Abstain ____

Not Participating ____ Request Discussion ____

COMMENTS:

I disapprove the staff proposal to develop a rulemaking plan for modifying 10 CFR 70.24 to address the long-term issue of recurring exemptions to the regulation. The staff should proceed with a direct final rulemaking, modifying 10 CFR 70.24, as appropriate. In addition, the staff should develop a generic methodology to address potential enforcement issues and exemption requests in the interim until the final rule becomes effective.

The staff should inform the Commission of the milestones and completion dates for this rulemaking.

Shirley Ann Jackson

Shirley Ann Jackson
SIGNATURE

Release Vote /__/

August 8, 1997
DATE

Withhold Vote /x/

Entered on "AS" Yes ____ No ____

# NOTATION VOTE

## RESPONSE SHEET

TO: John C. Hoyle, Secretary

FROM: COMMISSIONER DICUS

SUBJECT: SECY-97-155 - STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

Approved x Disapproved ____ Abstain ____

Not Participating ____ Request Discussion ____

COMMENTS:

See attached.

Greta Joy Dicus
SIGNATURE

August 13, 1997
DATE

Release Vote /__/

Withhold Vote /__/

Entered on "AS" Yes X No ____

**Comment on SECY-97-155**

Although the staff has a proposed action plan to alleviate the 10 CFR 70.24 exemption problem, the staff should expedite the proposed rulemaking as soon as practical. The rulemaking plan should ensure that the new rule is accomplished within one (1) calendar year.

# NOTATION VOTE

## RESPONSE SHEET

TO: John C. Hoyle, Secretary

FROM: COMMISSIONER DIAZ

SUBJECT: SECY-97-155 - STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

Approved x Disapproved ____ Abstain ____

Not Participating ____ Request Discussion ____

COMMENTS:

I approve the planned actions by the staff regarding the exemptions from 10 CFR 70.24. I also support the staff's plan related to enforcement actions.

The staff should expedite the rulemaking activity to eliminate the recurring exemptions to the requirements in 10 CFR 70.24. Since the issues are straight forward and the staff's resolution plan are clearly articulated in this paper, the staff should consider whether it is feasible to proceed directly to rulemaking and start developing the draft rule.

SIGNATURE

8-5-97

DATE

Release Vote /__/

Withhold Vote /__/

Entered on "AS" Yes ____ No ____

9710100180 970819
PDR COMMS NRCC
CORRESPONDENCE PDR

# NOTATION VOTE

## RESPONSE SHEET

TO: John C. Hoyle, Secretary

FROM: COMMISSIONER MCGAFFIGAN

SUBJECT: SECY-97-155 - STAFF'S ACTION REGARDING EXEMPTIONS FROM 10 CFR 70.24 FOR COMMERCIAL NUCLEAR POWER PLANTS

Approved x with comment Disapproved ____ Abstain ____

Not Participating ____ Request Discussion ____

COMMENTS:

I approve the proposed actions for the exemptions from 10 CFR 70.24 and the related staff proposal on enforcement actions for operating commercial nuclear power plants. I agree with the Chairman and Commissioner Diaz that the staff should expedite rulemaking. This issue is straight forward and the regulation has minimal safety significance for operating commercial nuclear power plants. The staff should proceed with direct final rulemaking. The staff should issue appropriate generic communication regarding compliance and enforcement for the interim period until the rule can be appropriately corrected.

Edward McGaffigan

SIGNATURE

8/12/97

DATE

Release Vote /x/

Withhold Vote /__/

Entered on "AS" Yes x No ____

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

TIME REQUESTED OCT 21 P2:02

October 10, 1997

NRC INFORMATION NOTICE 97-77: EXEMPTIONS FROM THE REQUIREMENTS OF SECTION 70.24 OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS

## Addressees

All holders of operating licenses for nuclear power reactors.

## Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about actions the staff plans to take regarding enforcement actions and the granting of exemptions from the requirements of Section 70.24 of Title 10 of the Code of Federal Regulations (10 CFR 70.24). This information notice does not transmit or imply any new or changed requirements or staff positions. No specific action or written response is required.

## Description of Circumstances

Regulations in 10 CFR 70.24 require that each licensee authorized to possess more than a small amount of special nuclear material maintain in each area in which such material is handled, used, or stored a criticality monitoring system that will energize alarm systems if accidental criticality occurs. The staff has issued approximately 20 notices of violation for failures by licensees to meet the provisions of 10 CFR 70.24. Since the issuance of these notices of violation, the staff has found it appropriate to exercise enforcement discretion pursuant to Section VII B.6 of the Enforcement Policy, NUREG-1600, " General Statement of Policy and Procedure for NRC Enforcement Actions."

## Discussion

As stated in a policy issue information paper, dated July 21, 1997, from the Executive Director for Operations, NRC, to the NRC Commissioners (SECY-97-155), the staff has determined that it is appropriate to exercise enforcement discretion in this case because the safety significance of the failure to meet 10 CFR 70.24 is minimal provided controls are in place to ensure compliance with general design criteria (GDC) 62. Also, enforcement discretion is appropriate because the NRC staff did not recognize the need for an exemption during the licensing process; because the NRC previously took a position on this matter, as reflected in its letter of May 11, 1988, to the Tennessee Valley Authority (NRC Accession No. 8902240029) concerning the lack of a need for an exemption at the Browns Ferry Nuclear Plant; and because of the staff's intent to embark on rulemaking to amend 10 CFR 70.24.

The staff intends to withdraw the previously issued violations. As specified in SECY-97-155, the staff does not intend to take further enforcement action for failure to meet 10 CFR 70.24 provided licensees obtain an exemption from this regulation before the next receipt of fresh fuel or before the next planned movement of fresh fuel. The criteria that the staff is using to evaluate exemptions from 10 CFR 70.24 are given in SECY-97-155 and are presented for information in the attachment to this notice.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the person listed below or the appropriate regional office.

BMatthews

for Jack W. Roe, Acting Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contact: G. Wunder, NRR
301-415-1494
E-mail: gfw@nrc.gov

Attachments:
1. Staff Criteria for Evaluating Exemptions from 10 CFR 70.24
2. List of Recently Issued NRC Information Notices

## STAFF CRITERIA FOR EVALUATING EXEMPTIONS FROM 10 CFR 70.24 AS STATED IN SECY-97-155

1. Plant procedures do not permit more than [1 PWR or 3 BWR] new fuel [assembly/assemblies] to be in transit between their associated shipping cask and dry storage rack at one time.

2. The k-effective of the fresh fuel storage racks filled with fuel of the maximum permissible U-235 enrichment and flooded with pure water does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

3. If optimum moderation of fuel in the fresh fuel storage racks occurs when the fresh fuel storage racks are not flooded, the k-effective corresponding to this optimum moderation does not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

4. The k-effective of spent fuel storage racks filled with fuel of the maximum permissible U-235 enrichment and filled with pure water does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

5. The quantity of forms of special nuclear material, other than nuclear fuel, that are stored on site in any given area is less than the quantity necessary for a critical mass.

6. Radiation monitors, as required by GDC 63, are provided in fuel storage and handling areas to detect excessive radiation levels and to initiate appropriate safety actions.

7. The maximum nominal U-235 enrichment is 5 wt percent.

## LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

| Information Notice No. | Subject | Date of Issuance | Issued to |
|---|---|---|---|
| 97-76 | Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times | 10/14/97 | All holders of OLs for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel |
| 97-75 | Enforcement Sanctions Issued as a Result of Deliberate Violations of NRC Requirements | 09/24/97 | All U.S. Nuclear Regulatory Commission licensees |
| 97-74 | Inadequate Oversight of Contractors During Sealant Injection Activities | 09/24/97 | All holders of OLs for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel |
| 97-73 | Fire Hazard in the Use of a Leak Sealant | 09/23/97 | All holders of OLs for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel |

OL = Operating License
CP = Construction Permit

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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# Rules and Regulations

Federal Register
Vol. 62, No. 232
Wednesday, December 3, 1997



## NUCLEAR REGULATORY COMMISSION

### 10 CFR Parts 50 and 70

**RIN 3150–AF87**

### Criticality Accident Requirements

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Direct final rule with opportunity to comment.

**SUMMARY:** The Nuclear Regulatory Commission (NRC) is amending its regulations to provide light-water nuclear power reactor licensees with greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area where the material is handled, used, or stored. This action is taken as a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

**EFFECTIVE DATE:** The final rule is effective February 17, 1998, unless significant adverse comments are received by January 2, 1998. If the effective date is delayed, timely notice will be published in the Federal Register.

**ADDRESSES:** Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff.

Hand deliver comments to 11555 Rockville Pike, Maryland, between 7:30 am and 4:15 pm on Federal workdays.

Copies of any comments received may be examined at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

For information on submitting comments electronically, see the discussion under Electronic Access in the Supplementary Information section.

**FOR FURTHER INFORMATION CONTACT:** Stan Turel, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–6234, e-mail spt@nrc.gov.

**SUPPLEMENTARY INFORMATION:**

### Background

The Nuclear Regulatory Commission (NRC) is amending its regulations to provide persons licensed to construct or operate light-water nuclear power reactors with the option of either meeting the criticality accident requirements of paragraph (a) of 10 CFR 70.24 in handling and storage areas for SNM, or electing to comply with certain requirements that would be incorporated into 10 CFR Part 50. These are generally the requirements that the NRC has used to grant specific exemptions to the requirements of 10 CFR 70.24. In addition, the NRC is revising the current text of the section relating to seeking specific exemptions from regulations in 10 CFR 70.24(d) which provided that a licensee could seek an exemption to all or part of 10 CFR 70.24 for good cause because it is redundant to 10 CFR 70.14(a). A modified 10 CFR 70.24(d) is being added to provide that the requirements in paragraph (a) through (c) of 10 CFR Part 70.24 do not apply to holders of a construction permit or operating license for a nuclear power reactor issued pursuant to 10 CFR Part 50, or combined licenses issued under 10 CFR Part 52, if the holders comply with the requirements of 10 CFR 50.68 (b).

The Commission's regulations in 10 CFR 70.24 require that each licensee authorized to possess more than a small amount of SNM maintain a criticality monitoring system "using gamma- or neutron-sensitive radiation detectors which will energize clearly audible alarm signals if accidental criticality occurs" in each area in which such material is handled, used, or stored. The regulation also specifies sensitivity requirements for these monitors and details the training that licensees must conduct in connection with criticality monitor alarms. The purpose of this section is to ensure that if a criticality were to occur during the handling of SNM, personnel would be alerted and would take appropriate action.

Most nuclear power plant licensees were granted exemptions from 10 CFR 70.24 during the construction of their plants as part of the 10 CFR Part 70 license issued to permit the receipt of the initial core. Generally, these exemptions were not explicitly renewed when the 10 CFR Part 50 operating license, which now contained the combined Part 50 and Part 70 authority, was issued. The requirements in 10 CFR 70.24 prescribe the attributes required of the monitoring and alarm system. Compliance with these requirements may be unnecessary for commercial power reactors where the conditions which could lead to a criticality event are so unlikely that the probability of occurrence of an inadvertent criticality is negligible. The NRC anticipated that the regulation might be unnecessary for some licensees and included in 10 CFR 70.24(d) an invitation to any licensee to seek an exemption to the entire section or part of the section for good cause. A large number of exemption requests have been submitted by power reactor licensees and approved by the NRC based on safety assessments which concluded that the likelihood of criticality was negligible. Because of the experience gained in processing these exemption requests, the NRC concluded that the regulations should be amended to provide this flexibility without requiring licensees to go through the exemption process.

### Discussion

At a commercial nuclear power plant, the reactor core, the fresh fuel delivery area, the fresh fuel storage area, the spent fuel pool, and the transit areas among these, are areas where amounts of SNM sufficient to cause a criticality exist. In addition, SNM may be found in laboratory and storage locations of these plants, but an inadvertent criticality is not considered credible in these areas due to the amount and configuration of the SNM. The SNM that could be assembled into a critical mass at a commercial nuclear power plant is only in the form of nuclear fuel. Nuclear power plant licensees have procedures and the plants have design features to prevent inadvertent criticality. The inadvertent criticality that 10 CFR 70.24 is intended to address could only occur during fuel-handling operations.

In contrast, at fuel fabrication facilities SNM is found and handled routinely in various configurations in addition to fuel. Although the handling of SNM at these facilities is controlled by procedures, the variety of forms of SNM and the frequency with which it is handled provides greater opportunity for an inadvertent criticality than at a nuclear power reactor.

At power reactor facilities with uranium fuel nominally enriched to no greater than five (5.0) percent by weight, the SNM in the fuel assemblies cannot go critical without both a critical configuration and the presence of a moderator. Further, the fresh fuel storage array and the spent fuel pool are in most cases designed to prevent inadvertent criticality, even in the presence of an optimal density of unborated moderator. Inadvertent criticality during fuel handling is precluded by limitations on the number of fuel assemblies permitted out of storage at the same time. In addition, General Design Criterion (GDC) 62 in Appendix A to 10 CFR Part 50 reinforces the prevention of criticality in fuel storage and handling through physical systems, processes, and safe geometrical configuration. Moreover, fuel handling at power reactor facilities occurs only under strict procedural control. Therefore, the NRC considers a fuel-handling accidental criticality at a commercial nuclear power plant to be extremely unlikely. The NRC believes the criticality monitoring requirements of 10 CFR 70.24 are unnecessary as long as design and administrative controls are maintained.

Because the NRC considers an inadvertent criticality to be unlikely at a nuclear power reactor, by this rulemaking it is granting nuclear power reactor licensees a choice—either meet the criticality monitoring requirements of 10 CFR 70.24 or in lieu of those criticality monitoring requirements meet certain criteria related to procedures, plant design, and fuel enrichment. These criteria are incorporated into section 50.68(b) of 10 CFR Part 50 by this direct final rule.

The three changes in the requirements are as follows:

(1) Section 50.68(a) provides that each holder of a construction permit or operating license for a nuclear power reactor issued under Part 50, or a combined license for a nuclear power reactor issued under Part 52 shall comply with either 10 CFR 70.24 or the seven requirements in section 50.68(b).

(2) Section 50.68(b) provides that each licensee as described in 50.68(a) shall comply with the seven listed requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24.

(3) The revised section 70.24(d) provides that the requirements in 10 CFR 70.24 (a) through (c) do not apply to holders of a construction permit or operating license for a nuclear power reactor issued pursuant to 10 CFR Part 50, or combined licenses issued under 10 CFR Part 52, if the holders comply with the requirements of paragraph (b) of 10 CFR 50.68.

## Procedural Background

Because NRC considers these amendments to its rules to be noncontroversial and routine, public comment on these amendments is unnecessary. The amendments to the rules will become effective on February 17, 1998. However, if the NRC receives significant adverse comments on the companion proposal published concurrently in the proposed rules section of this **Federal Register** by January 2, 1998, then the NRC will publish a document that withdraws this action and will address the comments received in response to the amendments. Such comments will be addressed in a subsequent final rule. The NRC will not initiate a second comment period on this action.

## Findings

Upon review of this rulemaking, that the changes and additions addressed by this rulemaking do not significantly affect the environmental cost-benefit balance that otherwise would justify the licensing of a light-water nuclear power reactor. The basis for this finding is that this rule is a codification of practices in place and does not significantly affect the cost-benefit balance for a light-water reactor.

## Metric Policy

On October 7, 1992, the Commission published its final Policy Statement on Metrication. According to that policy, after January 7, 1993, all new regulations and major amendments to existing regulations were to be presented in dual units. The new addition and amendment to the regulations contain no units.

## Environmental Impact: Categorical Exclusion

The NRC has determined that this proposed regulation is the type of action described in categorical exclusion 10 CFR 51.22(c)(3). Therefore neither an environmental impact statement nor an environmental assessment has been prepared for this proposed regulation.

## Electronic Access

You may also provide comments via the NRC's interactive rulemaking web site through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking site, contact Ms. Carol Gallagher, (301) 415-6215; e-mail CAG@nrc.gov.

## Paperwork Reduction Act Statement

This direct final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq*). Existing requirements were approved by the Office of Management and Budget, approval numbers 3150–0009 and 3150–0011.

## Public Protection Notification

If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

## Regulatory Analysis

The structure of the current 10 CFR 70.24 is overly broad and places burden on a licensee to identify those areas or operations at its facility where the requirements are unnecessary, and to request an exemption if the licensee has sufficient reason to be relieved from the requirements. This existing structure has the potential to result in a large number of recurring exemption requests.

To relieve the burden on power reactor licensees of applying for, and the burden on the staff of granting recurring exemptions, this amendment permits power reactor facilities with nominal fuel enrichments no greater than 5 weight percent U–235 to be excluded from the scope of 10 CFR 70.24, provided they meet specific requirements being added to 10 CFR Part 50. This amendment is a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

The only other viable option to this amendment is for the NRC to do nothing and allow the licensees to continue requesting exemptions. If nothing is done, the licensees will continue to incur the costs of submitting exemptions and NRC will incur the costs of reviewing them. Under this rule, an easing of burden on the part of licensees results by their not having to request exemptions. Similarly, the NRC will not need to review and evaluate these exemption requests, resulting in an easing of burden for the NRC.

This rule is not a mandatory requirement, but an easing of burden action which results in regulatory efficiency. Also, the rule does not impose any additional costs on licensees, has no negative impact on the public health and safety, but will provide certain licensees savings, and savings to the NRC as well. Hence, the rule is shown to be cost beneficial.

The foregoing constitutes the regulatory analysis for this final rule.

### Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule, if adopted, will not have a significant economic impact on a substantial number of small entities. This rule affects only the licensees of nuclear power plants. These licensees, companies that are dominant in their service areas, do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act, 5 U.S.C. 601, or the size standards adopted by the NRC (10 CFR 2.810).

### Backfit Analysis

The Commission has determined that a backfit analysis is not needed. This rule is a codification of practices in place by the NRC and is not a modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures of organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable NRC staff position (10 CFR Chapter I).

### Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a "major rule" and has verified this determination with the Office of Information and Regulatory Affairs, Office of Management and Budget.

### List of Subjects

*10 CFR Part 50*

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

*10 CFR Part 70*

Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

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, the National Environmental Policy Act of 1969, as amended, and 5 U.S.C. 553, the NRC is adopting the following amendments to 10 CFR Parts 50 and 70.

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

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

**Authority:** Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951, as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851). Sections 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.68 is added under the center heading "Issuance, Limitations, and Conditions of Licenses and Construction Permits" to read as follows:

**§ 50.68 Criticality accident requirements.**

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part, or a combined license for a nuclear power reactor issued under part 52 of this chapter shall comply with either 10 CFR 70.24 of this chapter or requirements in paragraph (b).

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures may not permit handling and transportation at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum permissible U-235 enrichment and flooded with pure water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum permissible U-235 enrichment and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible U-235 enrichment must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with pure water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible U-235 enrichment must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with pure water.

(5) The quantity of SNM, other than nuclear fuel stored on site, is less than the quantity necessary for a critical mass.

(6) Radiation monitors, as required by GDC 63, are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to no greater than five (5.0) percent by weight.

## PART 70—DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

1. The authority citation for 10 CFR Part 70 continues to read as follows:

Authority: Secs. 51, 53, 161, 182, 183, 68 Stat. 929, 930, 948, 953, 954, as amended, sec. 234, 83 Stat. 444, as amended, sec. 1701, 106 Stat. 2951, 2952, 2953 (42 U.S.C. 2071, 2073, 2201, 2232, 2233, 2282, 2297f); secs. 201, as amended, 202, 204, 206, 88 Stat. 1242, as amended, 1244, 1245, 1246, (42 U.S.C. 5841, 5842, 5845, 5846).

Sections 70.1(c) and 70.20a(b) also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161). Section 70.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 70.21(g) also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Section 70.31 also issued under sec. 57d, Pub. L. 93-377, 88 Stat. 475 (42 U.S.C. 2077). Sections 70.36 and 70.44 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234).

Section 70.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237). Section 70.62 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

2. In §70.24, paragraph (d) is revised to read as follows:

**§70.24 Criticality accident requirements.**

* * * * *

(d) The requirements in paragraph (a) through (c) of this section do not apply to holders of a construction permit or operating license for a nuclear power reactor issued pursuant to part 50 of this chapter, or combined licenses issued under part 52 of this chapter, if the holders comply with the requirements of paragraph (b) of 10 CFR 50.68 of this chapter.

Dated at Rockville, Maryland this 14th day of November, 1997.

For the Nuclear Regulatory Commission.

L. Joseph Callan,

*Executive Director for Operations.*

[FR Doc. 97-31733 Filed 12-2-97; 8:45 am]

BILLING CODE 7590-01-P

## DEPARTMENT OF TRANSPORTATION

## Federal Aviation Administration

## 14 CFR Part 39

[Docket No. 95-CE-99-AD; Amendment 39-10229; AD 96-24-17 R1]

RIN 2120-AA64

## Airworthiness Directives; The Don Luscombe Aviation History Foundation Models 8, 8A, 8B, 8C, 8D, 8E, 8F, T-8F Airplanes; Correction

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule; correction.

SUMMARY: This document clarifies information in airworthiness directive (AD) 96-24-17, which applies to Don Luscombe Aviation History Foundation (Luscombe) Models 8, 8A, 8B, 8C, 8D, 8E, 8F, T-8F airplanes. AD 96-24-17 currently requires installing new inspection holes, modifying the wing tip fairings, and inspecting the wing spars for intergranular corrosion. The actions specified in AD 96-24-17 are intended to prevent wing spar failure from intergranular corrosion, which could result in structural failure of the wings and loss of control of the airplane. The AD was published with an Appendix providing an alternative method of compliance. Since issuance of AD 96-24-17, the FAA has re-examined the Appendix and has determined that clarification of certain inspections procedures is needed. This action clarifies the procedures specified in the Appendix of AD 96-24-17.

DATES: Effective January 27, 1997.

The incorporation by reference of the Don Luscombe Aviation History Foundation Recommendation #2, dated December 15, 1993, revised November 21, 1995, as listed in the regulations, was previously approved by the Director of the Federal Register as of January 27, 1997 (61 FR 66900, December 19, 1996).

FOR FURTHER INFORMATION CONTACT: Mr. Sol Davis, Aerospace Engineer, Los Angeles Aircraft Certification Office, FAA, 3960 Paramount Boulevard, Lakewood, California 90712; telephone (562) 627-5233; facsimile (562) 627-5210.

SUPPLEMENTARY INFORMATION:

### Discussion

On November 25, 1996, the FAA issued AD 96-24-17, Amendment 39-9841 (61 FR 66900, December 19, 1996), which applies to Luscombe Models 8, 8A, 8B, 8C, 8D, 8E, 8F, T-8F airplanes. This AD currently requires installing a total of four additional wing inspection holes in the metal covered wings to assist in conducting a more thorough examination of the wing spars, modifying the wing tip fairing so that it is removable, and providing easier access to the interior of the wings. A one time inspection for intergranular corrosion is required for both metal covered and fabric covered wings on these Luscombe 8 series airplanes in the areas of the front and rear spar extrusions of the wing installations.

### Need for the Correction

AD 96-24-17 was published with an Appendix that provided an alternative method of compliance. The FAA has received reports that certain portions of the Appendix need clarification. Therefore, the FAA re-examined the procedures specified in the Appendix and has clarified items 2, 4, 6, 7, and 8, as well as clarifying a note regarding additional wing support.

### Correction of Publication

This document clarifies the Appendix to AD 96-24-17, and adds the AD as an amendment to §39.13 of the Federal Aviation Regulations (14 CFR 39.13).

The AD, as corrected, is being printed in its entirety for the convenience of affected operators. The effective date of the AD remains January 27, 1997, which is the effective date of the AD as originally issued.

Since this action only clarifies the Appendix instructions, it has no adverse economic impact and imposes no additional burden on any person. Therefore, the FAA has determined that prior notice and opportunity for public comment are unnecessary.

### List of Subjects in 14 CFR Part 39

Air transportation, Aircraft, Aviation safety, Safety.

### Adoption of the Correction

Accordingly, pursuant to the authority delegated to me by the Administrator, the Federal Aviation Administration amends part 39 of the Federal Aviation Regulations (14 CFR part 39) as follows:

## PART 39—AIRWORTHINESS DIRECTIVES

1. The authority citation for part 39 continues to read as follows:

Authority: 49 USC 106(g), 40113, 44701.

**§39.13 [Amended]**

2. Section 39.13, is amended by removing Airworthiness Directive (AD) 96-24-17, Amendment 39-9841 (61 FR

performed in accordance with Appendix VIII of Section XI, Division 1, 1995, Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code.

(2) [Reserved]

* * * * *

[5] For ASME Code Editions and Addenda issued prior to the Winter 1977 Addenda, the Code Edition and Addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the Winter 1977 addenda and subsequent editions and addenda the method for determining the applicable Code editions and addenda is contained in Paragraph NCA-1140 of Section III of the ASME Code.

* * * * *

[7] For purposes of this regulation the proposed IEEE-279 became "in effect" on August 30, 1968, and the revised issue IEEE-279-1971 became "in effect" on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the NRC Library, Two White Flint North, 11545, Rockville Pike, Rockville, Maryland 20852-2738.

* * * * *

Dated at Rockville, MD this 27th day of October 1997.

For the Nuclear Regulatory Commission.

**L. Joseph Callan,**

*Executive Director for Operations.*

[FR Doc. 97-31588 Filed 12-2-97; 8:45 am]

BILLING CODE 7590-01-P

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## NUCLEAR REGULATORY COMMISSION

### 10 CFR Parts 50 and 70

**RIN 3150-AF87**

### Criticality Accident Requirements

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Proposed rule.

**SUMMARY:** The Nuclear Regulatory Commission (NRC) is amending its regulations to provide light-water nuclear power reactor licensees with greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area where the material is handled, used, or stored. This action is taken as a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

**DATES:** Comments on the proposed rule must be received on or before January 2, 1998.

**ADDRESSES:** Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudication Staff. Hand deliver comments to 11555 Rockville Pike, Maryland, between 7:45 am and 4:15 pm on Federal workdays.

Copies of any comments received may be examined at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

For information on submitting comments electronically, see the discussion under Electronic Access in the Supplementary Information section.

**FOR FURTHER INFORMATION CONTACT:** Stan Turel, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone (301) 415-6234, e-mail spt@nrc.gov.

**SUPPLEMENTARY INFORMATION:** For additional information see the Direct Final Rule published in the rules section of this **Federal Register**.

### Procedural Background

Because NRC considers this action noncontroversial and routine, we are publishing this proposed rule concurrently as a direct final rule. The direct final rule will become effective on February 17, 1998. However, if the NRC receives significant adverse comments on the direct final rule by January 2, 1998, then the NRC will publish a document that withdraws the direct final rule. If the direct final rule is withdrawn, the NRC will address in a Final Rule the comments received in response to the proposed revisions in a subsequent final rule. Absent significant modifications to the proposed revisions requiring republication, the NRC will not initiate a second comment period for this action in the event the direct final rule is withdrawn.

### Electronic Access

You may also provide comments via the NRC's interactive rulemaking web site through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking site, contact Ms. Carol Gallagher, (301) 415-6215; e-mail CAG@nrc.gov.

### List of Subjects

*10 CFR Part 50*

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

*10 CFR Part 70*

Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

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, the National Environmental Policy Act of 1969, as amended, and 5 U.S.C. 553, the NRC is considering adopting the following amendments to 10 CFR Parts 50 and 70.

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

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

**Authority:** Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951, as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851). Sections 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.68 is added under the center heading "Issuance, Limitations, and Conditions of Licenses and Construction Permits" to read as follows:

**§ 50.68 Criticality accident requirements.**

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part, or a combined license for a nuclear power reactor issued under part 52 of this chapter shall comply with either 10 CFR 70.24 of this chapter or requirements in paragraph (b).

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures may not permit handling and transportation at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum permissible U–235 enrichment and flooded with pure water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum permissible U–235 enrichment and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible U–235 enrichment must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with pure water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible U–235 enrichment must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with pure water.

(5) The quantity of SNM, other than nuclear fuel stored on site, is less than the quantity necessary for a critical mass.

(6) Radiation monitors, as required by GDC 63, are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U–235 enrichment of the fresh fuel assemblies is limited to no greater than five (5.0) percent by weight.

## PART 70—DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

1. The authority citation for 10 CFR Part 70 continues to read as follows:

Authority: Secs. 51, 53, 161, 182, 183, 68 Stat. 929, 930, 948, 953, 954, as amended, sec. 234, 83 Stat. 444, as amended, sec. 1701, 106 Stat. 2951, 2952, 2953 (42 U.S.C. 2071, 2073, 2201, 2232, 2233, 2282, 2297f); secs. 201, as amended, 202, 204, 206, 88 Stat. 1242, as amended, 1244, 1245, 1246, (42 U.S.C. 5841, 5842, 5845, 5846).

Sections 70.1(c) and 70.20a(b) also issued under secs. 135, 141, Pub. L. 97–425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161). Section 70.7 also issued under Pub. L. 95–601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 70.21(g) also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Section 70.31 also issued under sec. 57d, Pub. L. 93–377, 88 Stat. 475 (42 U.S.C. 2077). Sections 70.36 and 70.44 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234).

Section 70.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237). Section 70.62 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

2. In § 70.24, paragraph (d) is revised to read as follows:

**§ 70.24 Criticality accident requirements.**

* * * * *

(d) The requirements in paragraph (a) through (c) of this section do not apply to holders of a construction permit or operating license for a nuclear power reactor issued pursuant to part 50 of this chapter, or combined licenses issued under part 52 of this chapter, if the holders comply with the requirements of paragraph (b) of 10 CFR 50.68 of this chapter.

Dated at Rockville, Maryland this 14th day of November, 1997.

For the Nuclear Regulatory Commission.

**L. Joseph Callan,**

*Executive Director for Operations.*

[FR Doc. 97–31732 Filed 12–2–97; 8:45 am]

BILLING CODE 7590–01–P

## DEPARTMENT OF TRANSPORTATION

### Federal Aviation Administration

### 14 CFR Part 39

[Docket No. 96–SW–22–AD]

### Airworthiness Directives; Eurocopter France (Formerly Aerospatiale, Society Nationale Industrielle, Sud Aviation) Model SA–365N, SA–365N1, AS–365N2, and SA–366G1 Helicopters

**AGENCY:** Federal Aviation Administration, DOT.

**ACTION:** Notice of proposed rulemaking (NPRM).

**SUMMARY:** This document proposes the adoption of a new airworthiness directive (AD) that is applicable to Eurocopter France (formerly Aerospatiale, Society Nationale Industrielle, Sud Aviation) Model SA–365N, SA–365N1, AS-365N2, and SA–366G1 helicopters. This proposal would require an inspection of the transmission deck for cracks; repair of any cracked transmission decks; and replacement of the transmission deck support beams (support beams) with redesigned support beams. This proposal is prompted by several reports of cracks in the transmission deck and support beams. The actions specified by the proposed AD are intended to detect cracks that reduce the strength of the main gearbox strut attachment and could result in failure of the main gearbox mounting, and subsequent loss of control of the helicopter.

**DATES:** Comments must be received by February 2, 1998.

**ADDRESSES:** Submit comments in triplicate to the Federal Aviation Administration (FAA), Office of the Regional Counsel, Southwest Region, Attention: Rules Docket No. 96–SW–22–AD, 2601 Meacham Blvd., Room 663, Fort Worth, Texas 76137. Comments may be inspected at this location between 9:00 a.m. and 3:00 p.m., Monday through Friday, except Federal holidays.

The service information referenced in the proposed rule may be obtained from American Eurocopter Corporation, 2701 Forum Drive, Grand Prairie, Texas 75053–4005, telephone (972) 641–3460, fax (972) 641–3527. This information may be examined at the FAA, Office of the Regional Counsel, Southwest Region, 2601 Meacham Blvd., Room 663, Fort Worth, Texas.

**FOR FURTHER INFORMATION CONTACT:** Mr. Mike Mathias, Aerospace Engineer, FAA, Rotorcraft Directorate, ASW–111, 2601 Meacham Blvd., Fort Worth, Texas

(5)

DOCKETED
USNRC

January 2, 1998

DOCKET NUMBER
PROPOSED RULE PR 50+70
(62FR63825)
(62FR63911)

'98 JAN -6 A1

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Secretary
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attn: Rulemaking and Adjudications Staff

The following comments are respectively submitted in response to the proposed changes to Criticality Accident Requirements, 10 CFR 50.68 and 70.24, published in Federal Register Volume 62, Number 232, Page 63825, December 3, 1997.

The phrase "as required by GDC 63" of proposed 10 CFR 50.68 (b) (6) should be removed for the following reasons. First, some plants were licensed before the General Design Criteria were promulgated and their licensing bases address the GDC on a case-by-case basis; the phrase in question infers that the General Design Criteria as stated in 10 CFR Part 50 Appendix A are part of every licensees' design basis. Second, the phrase does not add any substance since proposed 50.68 (b) (6) simply restates the relevant portion of GDC 63; omitting the reference would be consistent with proposed 50.68 (b) (1) through (5) which implement GDC 62 without specific reference to that GDC. Third, a person unfamiliar with 10 CFR 50 Appendix A would not recognize the reference to GDC 63 as stated.

Proposed 10 CFR 50.68 (b) (7), which places a five (5.0) weight percent limit on U-235 enrichment, should be eliminated and the phrase "maximum permissible U-235 enrichment" in proposed 50.68 (b) (2), (3), and (4) should be replaced by the phrase "maximum fuel assembly reactivity" for the following reasons. First, the discussion in the Federal Register announcement does not indicate that the enrichment limitation is the basis for a safety analysis; it is simply a statement of current practice. Second, the safety issue is fuel assembly reactivity of which enrichment is only one parameter; burnable poison, material selection, and geometry are major factors affecting reactivity that could compensate for higher enrichments. Third, by modifying 50.68 (b) (2), (3), and (4) as proposed, the reactivity limitation objective of fuel storage racks can be achieved without placing a limitation on fuel enrichment.

We appreciate the opportunity to comment on this proposed rule change.

Marcus H. Voth
Project Manager - Licensing
612-271-5116, marcus.h.voth@nspco

* Letter received by electronic mail on January 2, 1998 --- ATB

Northern States Power Company
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

(1) Employment in an excluded category follows employment subject to subchapter III of chapter 83 of title 5, United States Code, without a break in service or after a separation from service of 3 days or less, except in the case of:

(i) An alien employee whose duty station is located in a foreign country; or

(ii) An employee hired by the Census Bureau under a temporary, intermittent appointment to perform decennial census duties.

* * * * *

## PART 842—FEDERAL EMPLOYEES RETIREMENT SYSTEM—BASIC ANNUITY

3. The authority citation for part 842 is revised to read as follows:

**Authority:** 5 U.S.C. 8461(g); §§ 842.104 and 842.106 also issued under 5 U.S.C. 8461(n); § 842.105 also issued under 5 U.S.C. 8402(c)(1) and 7701(b)(2); § 842.106 also issued under section 102(e) of Pub. L. 104–8, 109 Stat. 102, as amended by section 153 of Pub. L. 104–134, 110 Stat. 1321; § 842.107 also issued under sections 11202(f), 11232(e), and 11246(b) of Pub. L. 105–33, 111 Stat. 251; §§ 842.604 and 842.611 also issued under 5 U.S.C. 8417; § 842.607 also issued under 5 U.S.C. 8416 and 8417; § 842.614 also issued under 5 U.S.C. 8419; § 842.615 also issued under 5 U.S.C. 8418; § 842.703 also issued under section 7001(a)(4) of Pub. L. 101–508; § 842.707 also issued under section 6001 of Pub. L. 100–203; § 842.708 also issued under section 4005 of Pub. L. 101–239 and section 7001 of Pub. L. 101–508; subpart H also issued under 5 U.S.C. 1104.

### Subpart A—Coverage

4. In § 842.105, paragraph (b) is revised to read as follows:

**§ 842.105 Regulatory exclusions.**

* * * * *

(b) When an employee who is covered by FERS moves to a position listed in paragraph (a) of this section without a break in service or after a separation of 3 days or less, his or her FERS coverage will continue, except in the case of an employee hired by the Census Bureau under a temporary, intermittent appointment to perform decennial census duties.

* * * * *

## PART 870—FEDERAL EMPLOYEES' GROUP LIFE INSURANCE PROGRAM

5. The authority citation for part 870 is revised to read as follows:

**Authority:** 5 U.S.C. 8716; subpart J also issued under sec. 599C of Pub. L. 101–513, 104 Stat. 2064, as amended; § 870.302 also issued under sections 11202(f), 11232(e), and 11246(b) and (c) of Pub. L. 105–33, 111 Stat. 251.

6. In § 870.301, add paragraph (c) to read as follows:

**§ 870.301 Eligibility for life insurance.**

* * * * *

(c) Notwithstanding any other provision in this part, the hiring of a Federal employee, whether in pay status or nonpay status, for a temporary, intermittent position with the decennial census has no effect on the amount of his/her Basic or Option B insurance, the withholdings or Government contribution for his/her insurance, or the determination of when 12 months in nonpay status ends.

## PART 890—FEDERAL EMPLOYEES HEALTH BENEFITS PROGRAM

7. The authority citation for part 890 continues to read as follows:

**Authority:** 5 U.S.C. 8913; § 890.803 also issued under 50 U.S.C. 403p, 22 U.S.C. 4069c and 4069c–1; subpart L also issued under sec. 599C of Pub. L. 101–513, 104 Stat. 2064, as amended; § 890.102 also issued under sections 11202(f), 11232(e), and 11246(b) and (c) of Pub. L. 105–33, 111 Stat. 251.

8. In § 890.102, paragraph (g) is added to read as follows:

**§ 890.102 Coverage.**

* * * * *

(g) Notwithstanding any other provision in this part, the hiring of a Federal employee, whether in pay status or nonpay status, for a temporary, intermittent position with the decennial census has no effect on the withholding or Government contribution for his/her coverage or the determination of when 365 days in nonpay status ends.

[FR Doc. 98–4781 Filed 2–24–98; 8:45 am]
BILLING CODE 6325–01–P

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## NUCLEAR REGULATORY COMMISSION

### 10 CFR Parts 50 and 70

RIN 3150–AF87

### Criticality Accident Requirements; Withdrawal of Direct Final Rule and Revocation of Regulatory Text

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Direct final rule; withdrawal.

**SUMMARY:** The Nuclear Regulatory Commission is withdrawing a direct final rule that would have amended the Commission's regulations to provide light-water nuclear power reactor licensees with greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area where the material is handled, used, or stored. The NRC is taking this action because it has received significant adverse comments in response to an identical proposed rule which was concurrently published in the **Federal Register**. Because the effective date for the direct final rule has passed, the NRC is removing the regulatory text codified in that action.

**EFFECTIVE DATE:** February 25, 1998.

**FOR FURTHER INFORMATION CONTACT:** Stan Turel, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415–6234 (E-mail: spt@nrc.gov).

**SUPPLEMENTARY INFORMATION:** On December 3, 1997 (62 FR 63825), the Nuclear Regulatory Commission published in the **Federal Register** a direct final rule amending its regulations to provide persons licensed to construct or operate light-water nuclear power reactors with the option of either meeting the criticality accident requirements of paragraph (a) of 10 CFR 70.24 in handling and storage areas for SNM, or electing to comply with requirements that would be incorporated into 10 CFR part 50 at § 50.68. The direct final rule was to become effective on February 17, 1998. The NRC also concurrently published an identical proposed rule on December 3, 1997 (62 FR 63911). In these documents, the NRC indicated that if it received significant adverse comments in response to this action, the NRC would withdraw the direct final rule and would consider the comments received as in response to the proposed rule and address these comments in a subsequent final rule. Therefore, the Commission is withdrawing the December 3, 1997, direct final rule. The public comments received will be addressed in a subsequent final rule issued in either a notice of final rulemaking or in a notice of withdrawal of the proposed rule.

Because this notice of withdrawal is being published after the February 17, 1998, effective date for the direct final rule, the regulatory text presented in the December 3, 1997, direct final rule must be removed from the Code of Federal Regulations. Therefore, the provisions added to part 50 at § 50.68 are removed and the text of § 70.24(d) is being restored to the text of the paragraph that was in effect before the December 3, 1997, amendment to that paragraph.

**List of Subjects**

*10 CFR Part 50*

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

*10 CFR Part 70*

Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

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 adopting the following amendments to 10 CFR parts 50 and 70.

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

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

Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955 as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, and 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80—50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C 2237).

### § 50.68 [Removed]

2. Section 50.68 is removed.

## PART 70—DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

3. The authority citation for part 70 continues to read as follows:

Authority: Secs. 51, 53, 161, 182, 183, 68 Stat. 929, 930, 948, 953, 954, as amended, sec. 234, 83 Stat. 444, as amended, (42 U.S.C. 2071, 2073, 2201, 2232, 2233, 2282, 2297f); secs. 201, as amended, 202, 204, 206, 88 Stat. 1242, as amended, 1244, 1245, 1246 (42 U.S.C. 5841, 5842, 5845, 5846). Sec. 193, 104 Stat. 2835 as amended by Pub. L. 104-134, 110 Stat. 1321, 1321-349 (42 U.S.C. 2243).

Sections 70.1(c) and 70.20a(b) also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161). Section 70.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 70.21(9) also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Section 70.31 also issued under sec. 57d, Pub. L. 93-377, 88 Stat. 475 (42 U.S.C. 2077). Sections 70.36 and 70.44 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 70.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237). Section 70.62 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

4. In § 70.24, paragraph (d) is revised to read as follows:

### § 70.24 Criticality accident requirements.

* * * * *

(d) Any licensee who believes that good cause exists why he should be granted an exemption in whole or in part from the requirements of this section may apply to the Commission for such exemption. Such application shall specify his reason for the relief requested.

Dated at Rockville, Maryland, this 20th day of February, 1998.

For the Nuclear Regulatory Commission.

**John C. Hoyle,**

*Secretary of the Commission.*

[FR Doc. 98-4830 Filed 2-24-98; 8:45 am]

BILLING CODE 7590-01-P

## DEPARTMENT OF TRANSPORTATION

## Federal Aviation Administration

## 14 CFR Part 39

[Docket No. 97-SW-29-AD; Amendment 39-10359; AD 98-04-48]

RIN 2120-AA64

## Airworthiness Directives; Eurocopter France Model AS 332L2 Helicopters

**AGENCY:** Federal Aviation Administration, DOT.

**ACTION:** Final rule; request for comments.

**SUMMARY:** This amendment adopts a new airworthiness directive (AD) that is applicable to Eurocopter France Model AS 332L2 helicopters. This action requires modifying the main rotor blade vibration absorber (vibration absorber) by replacing the weight support assemblies with reinforced weight support assemblies. This amendment is prompted by a report of the failure of a weight support assembly in-flight. The actions specified in this AD are intended to prevent failure of a vibration absorber weight support assembly, which could lead to adverse vibrations, contact between the fuselage and a main rotor blade or loss of a main rotor blade; and subsequent loss of control of the helicopter.

**DATES:** Effective March 12, 1998.

The incorporation by reference of certain publications listed in the regulations is approved by the Director of the Federal Register as of March 12, 1998.

Comments for inclusion in the Rules Docket must be received on or before April 27, 1998.

**ADDRESSES:** Submit comments in triplicate to the Federal Aviation Administration (FAA), Office of the Regional Counsel, Southwest Region, Attention: Rules Docket No. 97-SW-29-AD, 2601 Meacham Blvd., Room 663, Fort Worth, Texas 76137.

The service information referenced in this AD may be obtained from American Eurocopter Corporation, 2701 Forum Drive, Grand Prairie, Texas 75053-4005, telephone (972) 641-3460, fax (972) 641-3527. This information may be examined at the FAA, Office of the Regional Counsel, Southwest Region, 2601 Meacham Blvd., Room 663, Fort Worth, Texas; or at the Office of the Federal Register, 800 North Capitol Street, NW., suite 700, Washington, DC.

**FOR FURTHER INFORMATION CONTACT:** Mr. Mike Mathias, Aerospace Engineer, FAA, Rotorcraft Directorate, Rotorcraft Standards Staff, 2601 Meacham Blvd., Fort Worth, Texas 76137, telephone (817) 222-5123, fax (817) 222-5961.

**SUPPLEMENTARY INFORMATION:** The Direction Generale De L'Aviation Civile (DGAC), which is the airworthiness authority for France, recently notified the FAA that an unsafe condition may exist on Eurocopter France Model AS 332L2 helicopters with vibration absorbers, part number (P/N) 332A11-0460-01, installed. The DGAC advises that failure of a vibration absorber can result in adverse vibrations, contact between the fuselage and a main rotor blade or loss of a main rotor blade; and subsequent loss of control of the helicopter.

Eurocopter France has issued Eurocopter Service Bulletin No.

# Rules and Regulations

Federal Register
Vol. 63, No. 218
Thursday, November 12, 1998



## NUCLEAR REGULATORY COMMISSION

### 10 CFR Parts 50 and 70

**RIN 3150–AF87**

### Criticality Accident Requirements

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Final rule.

**SUMMARY:** The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to give licensees of light-water nuclear power reactors greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area in which the material is handled, used, or stored. This action is taken as a result of the experience gained in processing and evaluating a number of exemption requests from such licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

**EFFECTIVE DATE:** The final rule is effective on December 14, 1998.

**FOR FURTHER INFORMATION CONTACT:** Michael T. Jamgochian, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; telephone: (301) 415–3224; e-mail: mtj1@nrc.gov.

**SUPPLEMENTARY INFORMATION:**

## I. Background

The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to give persons licensed to construct or operate light-water nuclear power reactors the option of either meeting the criticality accident requirements of paragraph (a) through (c) of 10 CFR 70.24 in handling and storage areas for SNM, or electing to comply with certain requirements that are set forth in a new Section 50.68 in 10 CFR Part 50. The requirements in Section 50.68 are generally the requirements that the NRC has used to grant specific exemptions from the requirements of 10 CFR 70.24. In addition, the NRC is deleting the current text of Section 70.24(d) concerning the granting of specific exemptions from Section 70.24 because it is redundant to 10 CFR 70.14(a). Section 70.24(d) is rewritten to provide that the requirements in paragraphs (a) through (c) of 10 CFR 70.24 do not apply to holders of a construction permit or operating license for a nuclear power reactor issued under 10 CFR Part 50, or combined licenses issued under 10 CFR Part 52, if the holders comply with the requirements of 10 CFR 50.68(b).

## II. Discussion

On December 3, 1997 (62 FR 63825), the NRC published a direct final rule in the **Federal Register** that would have provided persons licensed to construct or operate light-water nuclear power reactors with the option of either meeting the criticality accident requirements of paragraph (a) of 10 CFR 70.24 in handling and storage areas for SNM, or electing to comply with requirements that would be incorporated into 10 CFR Part 50 at 10 CFR 50.68. A direct final rule (62 FR 63825) and a parallel proposed rule (62 FR 63911) amending Parts 70 and 50 were published in the **Federal Register** on December 3, 1997. The statement of considerations for the direct final rule and the proposed rule stated that if significant adverse comments were received on the direct final rule, the NRC would withdraw the direct final rule and would address the comments in a subsequent final rule. Significant adverse comments were received from the public, and on February 25, 1998, the NRC published a notice withdrawing the direct final rule and revoking the regulatory text. Since the direct final rule had an effective date of February 17, 1998, it was necessary for the February 25, 1998 notice to revoke the regulatory text which became effective on February 17, 1998, as well as to withdraw the direct final rule. With the withdrawal and revocation, the proposed rule is the only regulatory proposal remaining. The NRC has determined to modify the proposed rule to address public comments and to make several editorial clarifications. The analysis of and response to the public comments to the proposed rule are set forth below.

## III. Comments on the Proposed Rule

The NRC received comments on the December 3, 1997, proposed rule (62 FR 63911) from Commonwealth Edison, Carolina Power & Light Company, Southern Nuclear Operating Company, Nuclear Energy Institute, Northern States Power Company, Trojan Nuclear Plant, and Detroit Edison. Copies of the letters are available for public inspection and copying for a fee at the Commission's Public Document Room, located at 2120 L Street, NW. (Lower Level), Washington, DC. Many of the comment letters suggested editorial type changes, some of which have been incorporated into this final rule. The comments are classified into nine general comments and are addressed as follows:

*Comment 1:* The proposed rule should not prohibit licensees from applying for exemptions under the guidelines of 10 CFR 70.14 and should contain provisions to note that any existing approved exemptions remain valid.

*Response:* Even though the wording of paragraph (d) in the current version of 10 CFR 70.24, which provides for applying for exemptions should "good cause" exist, is being deleted, licensees are not prohibited from applying for such exemptions under the guidelines of paragraph (a) of 10 CFR 70.14, "Specific Exemptions."

The standard for issuance of exemptions under Section 70.14 is essentially the same as the "good cause" criterion in paragraph (d) of Section 70.24. Therefore, its removal from Section 70.24(d) will not change the standard for, or otherwise serve to limit the granting of, exemptions to Section 70.24.

This rulemaking does not affect the status of exemptions to the requirements of Section 70.24 that were previously granted by the NRC. A licensee currently holding an exemption to Section 70.24 may continue operation under its existing exemption (including any applicable conditions imposed as part of the granting of the exemption) and its current programs and commitments without any further action. Alternatively, a licensee currently holding exemptions to Section 70.24 may elect to comply with the new alternative provided under Section 50.68(b), but if it does so, its exemption would be inapplicable and would not serve as a basis for avoiding compliance with the criteria listed in Section 50.68(b). A licensee whose exemption was issued as part of its operating license and whose exemption contained conditions imposed as part of the granting of the exemption, need not apply for a license amendment to delete the exemption conditions as a prerequisite for complying with Section 50.68(b).

*Comment 2:* For many BWRs, optimum moderation calculations are not performed for the fresh fuel storage racks because administrative controls are in place to preclude these conditions. In accordance with vendor recommendations, compensatory measures have been established to preclude an optimum moderation condition in the fresh fuel storage racks. The rule should contain a provision that exempts this requirement if adequate controls have been established to preclude an optimum moderation condition.

*Response:* The NRC agrees and has added the following provision to 10 CFR 50.68(b)(3): "This evaluation need not be performed if administrative control and/or design features prevent such moderation, or if fresh fuel storage racks are not used."

*Comment 3.* The rule should eliminate the reference to General Design Criterion 63 (GDC 63) and should describe the underlying monitoring requirements.

*Response:* The reference to GDC 63 was initially incorporated to ensure that licensees receiving an exemption to 10 CFR 70.24 would not erroneously view the exemption as the basis for removing from the spent fuel pool area radiation monitors that were installed to meet other monitoring requirements, such as those contained in 10 CFR 20.1501 and GDC 63. This rule change does not affect these other monitoring requirements; therefore, referencing GDC 63 has been deleted.

*Comment 4.* Placing a limit on enrichment offers no direct safety benefit and should not be included.

*Response:* The NRC disagrees with the comment. The maximum allowable nominal enrichment of reactor fuel is currently limited to 5-weight percent on the basis of possible criticality concerns even in a dry environment, as well as currently approved extensions to 10 CFR 51.52 based on an environmental impact study for enrichments higher than 5-weight percent. Any future approved enrichment extension can be readily handled by modifying this criterion.

*Comment 5.* Replace "may not permit" with "shall prohibit the" in Criterion (1).

*Response:* The NRC agrees and has used the phrase suggested by the commenters.

*Comment 6.* Use of "pure water" and "unborated water" should be consistent.

*Response:* The NRC agrees. The final rule uses the term "unborated water."

*Comment 7.* Criteria (2) and (3) should not be applicable if the licensee does not use the fresh fuel storage racks.

*Response:* The NRC agrees and has added the following provision to 10 CFR 50.68 (b)(2) and (b)(3): "This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

*Comment 8.* The meaning of "transportation" in criterion (1) is unclear.

*Response:* The NRC agrees and has deleted the term.

*Comment 9.* The phrase "maximum permissible U–235 enrichment" in Criteria (2), (3), and (4) should be replaced by the phrase "maximum fuel assembly reactivity."

*Response:* The NRC agrees and has used the phrase suggested by the commenter.

## IV. Section-by-Section Analysis

*10 CFR 50.68*

Paragraph (a) of Section 50.68 allows a nuclear power plant licensee (including a holder of either a construction permit or a combined operating license) the option of complying with Section 70.24 (a) through (c), or complying with the requirements in paragraph (b) of Section 50.68. The corresponding provision in Section 70.24 is paragraph (d).

Paragraph (b) sets forth eight specific requirements which a licensee must comply with so long as it chooses under the provisions of Section 50.68 to avoid compliance with the requirements of Section 70.24 (a) through (c).

A licensee currently holding an exemption to Section 70.24 may elect to comply with the new alternative provided under Section 50.68, but if it does so, its exemption to Section 70.24 is inapplicable to, and would not serve as a basis for avoiding compliance with the eight criteria in Section 50.68(b).

*10 CFR 70.24*

Paragraph (d)(1) of Section 70.24 allows a nuclear power plant licensee (including a holder of either a construction permit or a combined operating license) the option of complying with Section 70.24 (a) through (c), or complying with the requirements in 10 CFR Section 50.68. This paragraph is the corresponding provision to Section 50.68(a).

Paragraph (d)(2) clarifies that the status of exemptions to the requirements of Section 70.24 that were previously granted by the NRC continue unaffected by this rulemaking. A licensee currently holding an exemption to Section 70.24 may continue operation under its existing exemption (including any applicable conditions imposed as part of the grant of the exemption) and its current programs and commitments without any further action.

A license that seeks an exemption from the requirements of Section 70.24 must meet the criteria for an exemption under Section 70.14. The standard for issuance of exemptions remains unchanged from the old rule, since the Commission regards the former "good cause" criterion under the previous version of Section 70.24(d) as being essentially the same as the standard for issuance of exemptions under Paragraph 70.14.

## V. Metric Policy

On October 7, 1992, the Commission published its final Policy Statement on Metrication. According to that policy, after January 7, 1993, all new regulations and major amendments to existing regulations were to be presented in dual units. The new addition and amendment to the regulations contain no units.

## VI. Finding of No Significant Environmental Impact

The NRC 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, would not be a major Federal action significantly affecting the quality of the human environment; and therefore, an environmental impact statement is not required. The final rule provides an alternative to existing requirements on criticality monitoring. The alternative method contained in the final rule in the new Section 50.68 represents a codification of the criteria currently used by the NRC for granting exemptions from the criticality monitoring requirements in 10 CFR 70.24(a). These criteria provide an acceptable alternative for assuring that there are no inadvertent criticality events of special nuclear material at nuclear power reactors, which is the purpose of the criticality monitoring requirements in Section 70.24(a). Experience over 15 years has demonstrated that the alternative criteria have been effective in preventing inadvertent criticality events, and the NRC concludes that as a matter of regulatory efficiency, there is no purpose to requiring licensees to apply for and obtain exemptions from requirements of Section 70.24(a) if they adhere to the alternative criteria in the new Section 50.68. Since the alternative contained in Section 50.68 provides an equally effective method for preventing inadvertent criticality events in nuclear power plants, the NRC concludes that the final rule will not have any significant impact on the quality of the human environment. Therefore, an environmental impact statement has not been prepared for this regulation. This discussion constitutes the environmental assessment for this rulemaking.

### VII. Paperwork Reduction Act Statement

This final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget, approval numbers 3150-0009 and 3150-0011.

### VIII. Public Protection Notification

If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

### IX. Regulatory Analysis

The current structure of the current 10 CFR 70.24 is overly broad and places a burden on a licensee to identify those areas or operations at its facility where the requirements are unnecessary, and to request an exemption if the licensee has sufficient reason to be relieved from the requirements. This existing structure has resulted in a large number of exemption requests.

To relieve the burden on power reactor licensees of applying for, and the burden on the NRC of granting exemptions, this amendment permits power reactor facilities with nominal fuel enrichments no greater than 5-weight percent of U-235 to be excluded from the scope of 10 CFR 70.24, provided they meet specific requirements being added to 10 CFR Part 50. This amendment is a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests which concluded that the likelihood of criticality was negligible.

The only other viable option to this amendment is for the NRC to make no changes and allow the licensees to continue requesting exemptions. If no changes are made, the licensees will continue to incur the costs of submitting exemptions and NRC will incur the costs of reviewing them. Under this rule, an easing of the burden on licensees results from not having to request exemptions. Similarly, the NRC's burden will be reduced by avoiding the need to review and evaluate these exemption requests.

This rule is not a mandatory requirement, but an easing of burden action which results in regulatory efficiency. Also, the rule does not impose any additional costs on existing licensees and has no negative impact on public health and safety, but will provide savings to future licensees, and may provide some reduction in burden to current licensees whose current exemption includes conditions which are more restrictive than the requirements in Section 50.68. There will also be savings in resources to the NRC as well. Hence, the rule is shown to be cost beneficial.

The foregoing constitutes the regulatory analysis for this final rule.

### X. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the NRC hereby certifies that this rule, if adopted, will not have a significant economic impact on a substantial number of small entities. This rule affects only the licensees of nuclear power plants. These licensee companies that are dominant in their service areas, do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act, 5 U.S.C. 601, or the size standards adopted by the NRC (10 CFR 2.810).

### XI. Backfit Analysis

The NRC has determined that this rule does not impose a backfit as defined in 10 CFR 50.109(a)(1), since it provides an alternative to existing requirements on criticality monitoring. Accordingly, the NRC has not prepared a backfit analysis for this rule.

### XII. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a "major rule" and has verified this determination with the Office of Information and Regulatory Affairs, Office of Management and Budget.

### List of Subjects

*10 CFR Part 50*

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

*10 CFR Part 70*

Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

For the reasons stated in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, the National Environmental Policy Act of 1969, as amended, and 5 U.S.C. 553, the NRC is adopting the following amendments to 10 CFR Parts 50 and 70:

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

The authority citation for 10 CFR part 50 continues to read as follows:

1. **Authority:** Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951, as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 and 50.81

also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.68 is added under the center heading "Issuance, Limitations, and Conditions of Licenses and Construction Permits" to read as follows:

**§ 50.68 Criticality accident requirements.**

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under Part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

(5) The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

(6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).

## PART 70—DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL

The authority citation for 10 CFR part 70 continues to read as follows:

1. Authority: Secs. 51, 53, 161, 182, 183, 68 Stat. 929, 930, 948, 953, 954, as amended, sec. 234, 83 Stat. 444, as amended, sec. 1701, 106 Stat. 2951, 2952, 2953 (42 U.S.C. 2071, 2073, 2201, 2232, 2233, 2282, 2297f); secs. 201, as amended, 202, 204, 206, 88 Stat. 1242, as amended, 1244, 1245, 1246, (42 U.S.C. 5841, 5842, 5845, 5846).

Sections 70.1(c) and 70.20a(b) also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161). Section 70.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 70.21(g) also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Section 70.31 also issued under sec. 57d, Pub. L. 93-377, 88 Stat. 475 (42 U.S.C. 2077). Sections 70.36 and 70.44 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234).

Section 70.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237). Section 70.62 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

2. In § 70.24, paragraph (d) is revised to read as follows:

**§ 70.24 Criticality accident requirements.**

* * * * *

(d)(1) The requirements in paragraphs (a) through (c) of this section do not apply to a holder of a construction permit or operating license for a nuclear power reactor issued under part 50 of this chapter or a combined license issued under part 52 of this chapter, if the holder complies with the requirements of paragraph (b) of 10 CFR 50.68.

(2) An exemption from § 70.24 held by a licensee who thereafter elects to comply with requirements of paragraph (b) of 10 CFR 50.68 does not exempt that licensee from complying with any of the requirements in § 50.68, but shall be ineffective so long as the licensee elects to comply with § 50.68.

Dated at Rockville, Maryland this 28th day of October, 1998.

For the Nuclear Regulatory Commission.

**William D. Travers,**

*Executive Director for Operations.*

[FR Doc. 98-30253 Filed 11-10-98; 8:45 am]

BILLING CODE 7590-01-P

## DEPARTMENT OF TRANSPORTATION

## Federal Aviation Administration

## 14 CFR Part 39

**[Docket No. 98-NM-217-AD; Amendment 39-10880; AD 98-23-13]**

**RIN 2120-AA64**

## Airworthiness Directives; British Aerospace Model Viscount 744, 745, 745D, and 810 Series Airplanes

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule.

SUMMARY: This amendment supersedes an existing airworthiness directive (AD), applicable to all British Aerospace Model Viscount 700, 800, and 810 series airplanes, that currently requires repetitive inspections to detect cracks and corrosion in the inboard and outboard engine nacelle structures on the wings; replacement of any cracked fittings and mating struts; and treatment or replacement of any corroded fittings or struts. This amendment requires repetitive inspections to detect cracking or corrosion of the eye end fittings of the outboard engine lower support or of the bore of the taper pin holes, and repair, if necessary. This amendment also limits the applicability of the existing AD. This amendment is prompted by reports of cracked and separated lower eye end fittings. The actions specified by this AD are intended to detect and correct cracking of the eye end fittings of the outboard engine lower support, which could result in reduced structural integrity of the engine nacelle support structures.

DATES: Effective December 17, 1998.

The incorporation by reference of certain publications listed in the regulations is approved by the Director nrc pdr

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 6 1979

MEMORANDUM FOR: R. Bernero, Assistant Director, Material Safety Standards Division of Engineering Standards

FROM: D. Eisenhut, Acting Director, Division of Operating Reactors

SUBJECT: RESPONSE TO REGULATORY GUIDE REVIEW REQUEST

In your memorandum of July 11, 1979 you requested our review of working paper E of Regulatory Guide 1.13, Revision 2, "Design objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

We have completed our review of this document and offer the attached comments.

Darrell Eisenhut
D. Eisenhut, Acting Director
Division of Operating Reactors

Contact:
E. Lantz, X27110

Enclosure:
As stated

cc w/enclosure:
D. Eisenhut
B. Grimes
G. Lainas
E. Adensam
E. Lantz

7908270

COMMENTS ON PROPOSED REVISION 2
TO REGULATORY GUIDE 1.13

1. On page 1.13-13 there is a typographical error in Section 4.4.

2. On page 1.13-15 Section 5.2 should read:

   "5.2 The presence of a soluble neutron absorber in pool water shall not normally be used in the evaluation of $k_s$. However, when calculating the effects of Condition IV faults realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies."

3. Page 1.13-16 the second paragraph should read, "Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, amount of burnable poison, U-235 depletion, burnable poison depletion, plutonium building, etc.

4. Page 1.13-16 the first sentence of the fourth and final paragraph should read, "The allowable U-235 depletion in spent fuels without burnable poison must not be set too high."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 10 1981

MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DRAFT 1, REGULATORY GUIDE 1.13, REVISION 2, "SPENT FUEL STORAGE FACILITY DESIGN BASIS"

Enclosed for initial review of the ACRS Regulatory Activities Subcommittee are 20 copies of Revision 2 to Regulatory Guide 1.13 (Enclosure 1) and 20 copies of the draft regulatory guide package (Enclosures 2 and 3).

The draft regulatory guide is a proposed revision to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," which is being revised to endorse ANSI N210-1976/ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." This draft guide also incorporates modifications to be consistent with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and revised SRP 9.1.4, "Light Load Fuel Handling Systems."

Since this draft is preliminary, additional staff efforts, including review and resolution of public comments, will be necessary prior to implementation of a regulatory position. ACRS Regulatory Activity Subcommittee comments and recommendations are requested on the proposed regulatory position.

Keith G Steyer for

Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

Enclosures:
1. Revision 2 to R.G. 1.13 (6/7/81)
2. Draft Value/Impact Assessment (2/12/81)
3. ANSI N210-1976/ANS 57.2

CONTACT: Carl S. Schulten
~~443-5910~~
443 7910

9109080403

CSchulten:clf
6/17/81

# DRAFT 1 OF REVISION 2 TO REGULATORY GUIDE 1.13

# SPENT FUEL STORAGE FACILITY DESIGN BASIS

## A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing this criterion.

## B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard which details minimum design requirements for light water reactor spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" by the American National Standards Institute on April 12, 1976.

These facilities must be designed to:

a. Prevent loss of water from the fuel pool that would uncover fuel.

b. Protect the spent fuel from mechanical damage.

c. Provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. Loss of Water from Storage Pool

Unless protective measures are taken, loss of water from a fuel storage pool could cause overheating of the spent fuel, resultant damage to fuel cladding integrity, and could result in a release of radioactive materials to the environment. Natural events, such as earthquakes or high winds, could damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity would alleviate these concerns.

Dropping of heavy loads, such as a 100-ton fuel cask, although of low probability, should be considered in plant arrangements where such loads are positioned or moved in or over the spent fuel pool. Cranes which are capable of carrying heavy loads should be prevented, preferably by design rather than interlocks, from moving into the vicinity of the pool.

The negative pressure in the fuel handling building during movement of spent fuel should be at least minus 3.2 mm (-0.125 inches) water gauge to prevent exfiltration and to assure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment.

Even if the measures described above which are used to maintain the desired negative pressure are followed, small leaks from the building may still occur as a result of structural failure or other unforeseen events. For example, equipment failures in systems connected to the pool could result in loss of water from the pool if this loss is not prevented by design. A permanent fuel-pool-coolant makeup system with a moderate capability, and with suitable redundancy or backup, could prevent the fuel from being uncovered if these leaks should occur. Early detection of pool leakage and fuel damage could be provided by both pool-water-level monitors and radiation monitors. Both types of monitors should be designed to alarm both locally and in a continuously manned location. Timely operation of building filtration systems can be assured if these systems are actuated by a signal from local radiation monitors.

2. Mechanical Damage to Fuel

The release of radioactive material from fuel may occur during the refueling process, and at other times, as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or the dropping of objects onto fuel elements.

Missiles generated by high winds are also a potential cause of mechanical damage to fuel. This concern could be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. Limiting Potential Offsite Exposures

A relatively small amount of mechanical damage to the fuel or fuel overheating might cause significant offsite doses of radiation if no dose reduction features are provided. Use of a controlled leakage building surrounding the fuel storage pool, with associated capability to limit releases of radioactive material resulting from a refueling accident, would appear feasible and do much to eliminate this concern.

For the spent fuel pool cooling, makeup and cleanup systems, the staff will consider the design acceptable if it includes seismic Category 1 and tornado protection for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the storage building's filtration-ventilation systems. The pool building's filtration-ventilation systems should be designed to meet the guidelines of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

In all activities involving personnel exposure to radiation, attention should be directed toward keeping occupational radiation as low as reasonably achievable (ALARA). Efforts toward maintaining exposures ALARA should be included in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

## C. REGULATORY POSITION

The requirements that are included in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations"[1] are generally acceptable to the NRC staff. The staff has determined that this standard provides an adequate basis for complying with the requirements of General Design Criterion 61 "Fuel Storage and Handling and Radioactivity Control" of Appendix A "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50 as related to light water reactors and subject to the following clarifications and modifications:

1. The example in Section 4.2.4.3(1) should be modified. The inventory of radioactive materials that could possibly leak from the spent fuel building should correspond to the amount predicted to leak under the postulated maximum damage conditions resulting from the dropping of a spent fuel assembly in the spent fuel building. However, in any event, the inventory should not be less than the amount available due to rupture of all fuel rods of a spent fuel assembly. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."[2]

[1]Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

[2]Copies of Regulatory Guides may be obtained from the U.S. Nuclear Regulatory Commisson, Washington, D.C. 20555.

2. In addition to meeting the requirements of Section 5.1.12 the maximum potential kinetic energy capable of being developed by those objects handled above stored spent fuel, if dropped, is not to exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

3. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

4. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to keep tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to keep missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado missiles or from seismic damage to assure that nothing bypasses the ESF grade filtration system in the containment building. In the event an earthquake or a tornado missile damages both the fuel pool containment and the fuel pool cooling system, no credit can be given to the filtration system used to reduce the amount of airborne radioactivity.

5. In addition to meeting the requirements of Section 5.1.5.3, provisions should be made for handling highly radioactive non-fuel irradiated components in fuel pools. Either the design of the retrieval system or

administrative controls should be included which would prohibit unknowing retrieval of irradiated components.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface between the cask venting system and the installed building ventilation system should be provided. This interface would provide for the proper handling of the "vent-gas" generated from filling a dry, loaded cask with water and thereby minimizing personnel exposure from the untreated off gas.

7. In order to limit the potential offsite release of radioactivity during a Condition IV fuel handling accident, Section 5.3.3 should include the requirement that the released radioactivity be either contained or removed by filtration so that the dose to an individual is less than 10 CFR Part 100 guidelines. The calculated offsite dose to an individual from such an event should be well within (approximately 25% of) the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to assure that released activity does not bypass the filtration system, the engineered safety feature fuel storage building ventilation should provide and maintain a negative pressure of at least minus 3.2mm (-0.125 inches), water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed such that travel directly over the spent fuel storage pool or safety-related equipment is not possible. This should be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I and safety-related structures and equipment should be subject to a quality assurance program which meets the applicable provisions of Appendix B to 10 CFR Part 50. Further, those programs should obtain guidance from Regulatory Guide 1.28 endorsing ANSI N45.2 "Quality Assurance Program Requirements for Nuclear Facilities" and the applicable provisions of ANSI N45.2 daughter standards endorsed by Regulatory Guides.

The Regulatory Guides endorsing the applicable ANSI N45.2 daughter standards are as follows:

1.30 Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (N45.2.4).

1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (N45.2.2).

1.58 Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (N45.2.6).

1.64 Quality Assurance Requirements for the Design of Nuclear Power Plants (N45.2.11).

1.74 Quality Assurance Terms and Definitions (N45.2.10).

1.88 Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (N45.2.9).

1.94 Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (N45.2.5).

1.116 Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (N45.2.8).

1.123 Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (N45.2.13).

10. The spent fuel pool water temperature of 65.6°C (150°F) stated in Section 6.6.1(2)(a) exceeds the NRC staff recommended limit. With the normal cooling system in operation, the pool water temperature should be kept at or below 60°C (140°F) with full core offload except when the pool water temperature is based on comparative analyses of the pool conditions that have been found acceptable previously. The spent fuel pool water temperature recommended limits for normal and abnormal cases are indicated in the table below.

NORMAL OPERATION

| Case I | Case II |
|---|---|
| . both trains operational | . both trains operational |
| . normal refueling | . full core offload |
| . pool full of spent fuel | . pool full of spent fuel |
| Maximum operating temperature | Maximum operating temperature |
| < 48.9°C (120 °F) | < 60°C (140° F) |
| based on fogging criteria and personnel comfort | to protect the ion exchange resin from degradation |

ABNORMAL OPERATION

| Case III | Case IV |
| --- | --- |
| . one train operational | . no cooling loops operational |
| . normal refueling | . full core offload |
| . pool full of spent fuel | . pool full of spent fuel |
| Maximum operating temperature <60°C (140°F) | Pool boiling permitted |

11. A nuclear criticality safety analysis should be performed in accordance with Annex A for each light water reactor spent fuel storage facility that involves the handling, transfer, or storage of spent fuel assemblies.

12. Sections 6.4 and 9 of ANS 57.2 lists codes and standards that are referenced in this standard. Endorsement of ANS 57.2 by this regulatory guide does not constitute an endorsement of the referenced codes and standards.

## D. IMPLEMENTATION

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

This guide reflects current NRC staff practice for construction permit review. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission regulations, the methods described herein will be used in the evaluation of license applications docketed after ________________.

ANNEX A

## Nuclear Criticality Safety

1. Scope of Nuclear Criticality Safety Assessment

1.1 A nuclear criticality safety analysis shall be performed for each light water reactor spent fuel storage facility system that involves the handling, transfer, or storage of spent fuel assemblies.

1.2 The nuclear criticality safety analysis shall demonstrate that each reactor spent fuel storage facility system is subcritical ($k_{eff}$ shall not exceed 0.95).

1.3 The nuclear criticality safety analysis shall include consideration of all credible normal and abnormal operating occurrences, including:

a) Accidental tipping or falling of a spent fuel assembly

b) Accidental tipping or falling of a storage rack during transfer

c) Misplacement of a spent fuel assembly

d) Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system.

e) Fuel drop accidents

f) Stuck fuel assembly/crane uplifting forces

g) Horizontal motion of fuel before complete removal from rack

h) Placing a fuel assembly along the outside of rack

i) Objects that may fall onto the stored spent fuel assemblies

1.4 At all locations in the reactor spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis shall demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

1.5 The nuclear criticality safety analysis shall explicitly identify spent fuel assembly characteristics upon which subcriticality in the reactor spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis shall explicitly identify design limits upon which subcriticality depends that require physical verfication at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis shall explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

2. Calculational Methods and Codes

Methods used to calculate subcriticality shall be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety." (Endorses ANSI N16.9-1975)

## 3. Method to Establish Subcriticality

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks under normal and credible abnormal conditions shall be equal to or less than an established maximum allowable multiplication factor $k_a$; i.e.,

$$k_s \leq k_a \qquad \text{(Eq. 1)}$$

where

$k_s$ = the evaluated maximum multiplicaton factor of fuel in the spent fuel storage racks, including any necessary allowance for statistical uncertainties in the calculational technique such as in Monte Carlo calculations.

The maximum allowable multiplication factor shall be calculated from the expression:

$$k_a = k_c - \Delta k_u - \Delta k_m \qquad \text{(Eq. 2)}$$

where

$k_c$ = $k_{eff}$ computed for the most reactive fuel assembly at the most reactive point by the same calculational method which was used for the benchmark experiments.

<u>Note</u>: $k_c$ is the value of $k_{eff}$ that results from the calculation of the benchmark experiments using a particular calculational method. The value represents a combination of theoretical technique and numerical data. (For more detail, see Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety.")

$\Delta k_u$ = The uncertainty in the benchmark experiments.

$\Delta k_m$ = The value required to assure an accepted margin of subcriticality.

3.2 $\Delta k_u$ shall include both uncertainties in the benchmark experiments as well as uncertainties in the bias which result from extrapolation of the benchmark experiments into the range of parameters encountered in the spent fuel storage rack design.

3.3 $\Delta k_m$ shall provide an adequate margin of subcriticality under the operating limitations and Design Events I through IV, and shall be no less than 0.02 (new fuel).

3.4 In the absence of information that justifies a smaller margin of subcriticality, value of 0.05 shall be assumed for $\Delta k_m$ for the design of spent fuel storage racks (spent fuel).

4. Storage Rack Analysis Assumptions

4.1 The fuel assembly assumed for storage facility design shall be one of the following:

a) the most reactive fuel assembly to be stored, at the most reactive point in the fuel assembly's life with no allowance for fission product content due to burn-up; or

b) The most reactive fuel assembly to be stored based on a minimum confirmed burnup. If credit is taken for burnup, an allowable fuel assembly reactivity shall be established and it shall be shown by actual measurement that each fuel assembly meets this criterion before it is allowed to be placed in storage. (See Annex B.)

4.2 Determination of the most reactive spent fuel assembly shall include consideration of the following parameters:

- maximum fissile fuel loading,
- fuel rod diameter,
- fuel rod cladding material and thickness,
- fuel pellet density,
- fuel rod pitch and total number of fuel rods within assembly,
- absence of fuel rods in certain locations, and
- burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design shall be the arrangement that results in the highest value of $k_s$ considering:

a) spacing between assemblies,
b) moderation between assemblies, and
c) fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of $k_s$ shall include consideration of the following:

a) eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,

b) dimensional tolerances,

c) construction materials,

d) fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e) presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f) presence of structural material and fixed neutron absorber in cell walls between assemblies.

## 5. Use of Neutron Absorbers in Storage Rack Design

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a) The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b) Fixed neutron absorbers shall be an integral, non-removable part of the storage rack.

c) When a fixed neutron absorber is used as the primary nuclear criticality safety control, there shall be provision to:

   1) initially confirm absorber presence in the storage rack, and

2) periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water shall not normally be used in the evaluation of $k_s$. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

## ANNEX B

### Most Reactive Fuel Assembly to be Stored Based on a Minimum Confirmed Burnup

If credit is to be taken for fuel burnup in the design of spent fuel storage racks, an acceptable basis for setting and meeting the limit must be established. The rationale for this basis will evolve from many rather complex considerations.

Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, $^{235}U$ depletion, amount of burnable poison, plutonium buildin and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildin are not necessarily the same.

Consideration should be given to how burnup limits are selected and specified for a particular fuel type:

The allowable $^{235}U$ depletion in the spent fuels without burnable poison must not be set too high. If too much depletion is credited in the analysis compared to the range of $^{235}U$ depletion in spent fuel assemblies to be stored, the design could be nonconservative from the standpoint of criticality safety. On the other hand, if too little depletion is credited in the analysis compared to the spent fuel to be stored, then the design will be conservative. Thus a <u>maximum</u> depletion to be allowed in <u>design</u>

can be established consistent with the range of $^{235}U$ depletions expected in the spent fuel assemblies to be stored. (This limit would then correspond to the minimum depletion that would be allowed in a particular fuel assembly type destined to be stored in the racks.)

The allowable plutonium content in the spent fuel upon which design would be based must not be set too low. If design is based on too little plutonium compared to the range of plutonium concentrations that may be in the spent fuel assemblies to be stored in the racks, the design could be non-conservative from the standpoint of nuclear criticality safety. On the other hand, if too much plutonium is credited in the analysis of the storage racks compared to the spent fuel assemblies to be stored, then the design would be conservative. Thus, a minimum plutonium content to be allowed in design can be established consistent with the range of plutonium concentrations expected in the spent fuel assemblies to be stored.

(This limit would then correspond to the maximum plutonium content that would be allowed in a particular fuel assembly type destined to be stored in the racks.)

Credit for fission product content presents special problems, such as the identities and quantities of the various fission products present and how to evaluate the effect of decay rates on the credit taken. The allowable fission product content in the spent fuel upon which design would be based must not be set too high. If design is based on too high of a fission product content compared to the range of fission product concentrations that may be in the spent fuel assemblies to be stored in the racks, the design could be non-conservative from the standpoint of criticality safety. On the other hand, if too few fission products are credited in the analysis of the racks compared to the spent fuel assemblies to be stored, then the design would be conservative. Thus, with proper consideration a maximum fission product content to be allowed in design could be established consistent with the range of fission product concentrations expected in the spent fuel to be stored.

(This limit would then correspond to the minimum fission product content that would be allowed in a particular fuel assembly type to be stored in the racks.

Finally, consideration should be given to the practical implementation of the spent fuel screening process. Depletion of $^{235}U$ and plutonium and fission product buildin cannot be easily or practically determined analytically. An obvious approach would be to translate the allowable burnup to a net allowable fuel assembly reactivity and then measure every fuel assembly to confirm that the minimum criterion is met.

# VALUE/IMPACT ASSESSMENT ON NUCLEAR POWER PLANT SPENT FUEL STORAGE FACILITY DESIGN

## 1. PROPOSED ACTION

### 1.1 Description

Each nuclear power plant has a spent fuel storage facility. General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

### 1.2 Need for Proposed Action

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested this guide be updated.

### 1.3 Value/Impact of Proposed Action

#### 1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit application. Therefore, there should be a minimum of cases where the applicant and the staff radically disagree on the design criteria.

1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also recommendations made by Task A-36 which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" would also be included.

3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

## 4. STATUTORY CONSIDERATIONS

### 4.1 NRC AUTHORITY

This guide would fall under the authority and safety requirements of the Atomic Energy Act of 1954, as amended. In particular under General Design Criterion 61, Appendix A, 10 CFR Part 50 of the NRC's implementing regulations.

### 4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by 10 CFR Part 51.5(a)(10) and does not require an environmental impact statement.

## 5. CONCLUSION

Regulatory Guide 1.13 should be updated.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

U.S. NUCLEAR REGULATORY
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MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safety

FROM: Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DRAFT 1, REGULATORY GUIDE 1.13, REVISION 2, "SPENT FUEL STORAGE FACILITY DESIGN BASIS"

Enclosed for initial review of the ACRS Regulatory Activities Subcommittee are 20 copies of Revision 2 to Regulatory Guide 1.13 (Enclosure 1) and 20 copies of the Draft Value/Impact Assessment (Enclosure 2).

The draft regulatory guide is a proposed revision to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," which is being revised to endorse ANSI N210-1976/ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

The draft regulatory guide, which was originally scheduled for review at the September 9th meeting, was withdrawn to insure the incorporation of all necessary input from Division Offices.

Since this draft is preliminary, additional staff efforts, including review and resolution of public comments, will be necessary prior to implementation of a regulatory position. ACRS Regulatory Activities Subcommittee comments and recommendations are requested on the proposed regulatory position.

Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

cc: Public Document Room

Enclosures: as stated

# DRAFT 1 OF REVISION 2 TO REGULATORY GUIDE 1.13

# SPENT FUEL STORAGE FACILITY DESIGN BASIS

## A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing this criterion.

## B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard which details minimum design requirements for 10 CFR Part 50 light water reactor spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor

Spent Fuel Storage Facilities at Nuclear Power Stations" by the American National Standards Institute on April 12, 1976.

These facilities must be designed to:

a. Prevent loss of water from the fuel pool that would uncover fuel.

b. Protect the spent fuel from mechanical damage.

c. Provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. Loss of Water from Storage Pool

Unless protective measures are taken, loss of water from a fuel storage pool could cause overheating of the spent fuel, resultant damage to fuel cladding integrity, and could result in a release of radioactive materials to the environment. Natural events, such as earthquakes or high winds, could damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity would alleviate these concerns.

Dropping of heavy loads, such as a 100-ton fuel cask, although of low probability, should be considered in plant arrangements where such loads are positioned or moved in or over the spent fuel pool. Cranes which are capable of carrying heavy loads should be prevented, preferably by design rather than interlocks, from moving into the vicinity of the pool.

The negative pressure in the fuel handling building during movement of spent fuel should be at least minus 3.2 mm (-0.125 inches) water gauge to prevent exfiltration and to assure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment.

Even if the measures described above which are used to maintain the desired negative pressure are followed, small leaks from the building may still occur as a result of structural failure or other unforeseen events. For example, equipment failures in systems connected to the pool could result in loss of water from the pool if this loss is not prevented by design. A permanent fuel-pool-coolant makeup system with a moderate capability, and with suitable redundancy or backup, could prevent the fuel from being uncovered if these leaks should occur. Early detection of pool leakage and fuel damage could be provided by both pool-water-level monitors and radiation monitors. Both types of monitors should be designed to alarm both locally and in a continuously manned location. Timely operation of building filtration systems can be assured if these systems are actuated by a signal from local radiation monitors.

2. Mechanical Damage to Fuel

The release of radioactive material from fuel may occur during the refueling process, and at other times, as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or the dropping of objects onto fuel elements.

Missiles generated by high winds are also a potential cause of mechanical damage to fuel. This concern could be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. Limiting Potential Offsite Exposures

A relatively small amount of mechanical damage to the fuel or fuel overheating might cause significant offsite doses of radiation if no dose reduction features are provided. Use of a controlled leakage building surrounding the fuel storage pool, with associated capability to limit releases of radioactive material resulting from a refueling accident, would appear feasible and do much to eliminate this concern.

For the spent fuel pool cooling, makeup and cleanup systems, the staff will consider the design acceptable if it includes seismic Category 1 and tornado protection for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the storage building's filtration-ventilation systems. The pool building's filtration-ventilation systems should be designed to meet the guidelines of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

In all activities involving personnel exposure to radiation, attention should be directed toward keeping occupational radiation as low as reasonably achievable (ALARA). Efforts toward maintaining exposures ALARA should be included in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

## C. REGULATORY POSITION

The requirements that are included in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations"[1] are generally acceptable to the NRC staff. The staff has determined that this standard provides an adequate basis for complying with the requirements of General Design Criterion 61 "Fuel Storage and Handling and Radioactivity Control" of Appendix A "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50 as related to light water reactors and subject to the following clarifications and modifications:

1. The example in Section 4.2.4.3(1) should be modified. The inventory of radioactive materials that could possibly leak from the spent fuel building should correspond to the amount predicted to leak under the postulated maximum damage conditions resulting from the dropping of a spent fuel assembly in the spent fuel building. However, in any event, the inventory should not be less than the amount available due to rupture of all fuel rods of a spent fuel assembly. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."[2]

2. In addition to meeting the requirements of Section 5.1.12 the maximum potential kinetic energy capable of being developed by those objects handled

[1]Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

[2]Copies of Regulatory Guides may be obtained from the U.S. Nuclear Regulatory Commisson, Washington, D.C. 20555.

above stored spent fuel, if dropped, is not to exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

3. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

4. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to keep tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to keep missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado missiles or from seismic damage to assure that nothing bypasses the ESF grade filtration system in the containment building. In the event an earthquake or a tornado missile damages both the fuel pool containment and the fuel pool cooling system, no credit can be given to the filtration system used to reduce the amount of airborne radioactivity.

5. In addition to meeting the requirements of Section 5.1.5.3, provisions should be made for handling highly radioactive non-fuel irradiated components in fuel pools. Either the design of the retrieval system or administrative controls should be included which would prohibit unknowing retrieval of irradiated components.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface between the cask venting system and the installed building ventilation system should be provided. This interface would provide for the proper handling of the "vent-gas" generated from filling a dry, loaded cask with water and thereby minimizing personnel exposure from the untreated off gas.

7. In order to limit the potential offsite release of radioactivity during a Condition IV fuel handling accident, Section 5.3.3 should include the requirement that the released radioactivity be either contained or removed by filtration so that the dose to an individual is less than 10 CFR Part 100 guidelines. The calculated offsite dose to an individual from such an event should be well within (approximately 25% of) the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to assure that released activity does not bypass the filtration system, the engineered safety feature fuel storage building ventilation should provide and maintain a negative pressure of at least minus 3.2mm (-0.125 inches), water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed such that travel directly over the spent fuel storage pool or safety-related equipment is not possible. This should be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I and safety-related structures and equipment should be subject to a quality assurance program which meets the applicable provisions of Appendix B to 10 CFR Part 50. Further, those programs should obtain guidance from Regulatory Guide 1.28 endorsing ANSI N45.2 "Quality Assurance Program Requirements for Nuclear Facilities" and the applicable provisions of ANSI N45.2 daughter standards endorsed by Regulatory Guides.

The Regulatory Guides endorsing the applicable ANSI N45.2 daughter standards are as follows:

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1.94 Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (N45.2.5).

1.116 Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (N45.2.8).

1.123 Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (N45.2.13).

10. The spent fuel pool water temperature of 65.6°C (150°F) stated in Section 6.6.1(2)(a) exceeds the NRC staff recommended limit. With the normal cooling system in operation, the pool water temperature should be kept at or below 60°C (140°F) with full core offload except when the pool water temperature is based on comparative analyses of the pool conditions that have been found acceptable previously. The spent fuel pool water temperature recommended limits for normal and abnormal cases are indicated in the table below.

NORMAL OPERATION

| Case I | Case II |
|---|---|
| . both trains operational | . both trains operational |
| . normal refueling | . full core offload |
| . pool full of spent fuel | . pool full of spent fuel |
| Maximum operating temperature | Maximum operating temperature |
| < 48.9°C (120 °F) | < 60°C (140° F) |
| based on fogging criteria and personnel comfort | to protect the ion exchange resin from degradation |

ABNORMAL OPERATION

| Case III | Case IV |
|---|---|
| . one train operational | . no cooling loops operational |
| . normal refueling | . full core offload |
| . pool full of spent fuel | . pool full of spent fuel |
| Maximum operating temperature | Pool boiling permitted |
| <60°C (140°F) | |

11. A nuclear criticality safety analysis should be performed in accordance with Annex A for each light water reactor spent fuel storage facility that involves the handling, transfer, or storage of spent fuel assemblies.

12. Sections 6.4 and 9 of ANS 57.2 lists codes and standards that are referenced in this standard. Endorsement of ANS 57.2 by this regulatory guide does not constitute an endorsement of the referenced codes and standards.

## D. IMPLEMENTATION

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

This guide reflects current NRC staff practice for construction permit review. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission regulations, the methods described herein will be used in the evaluation of license applications docketed after ______________.

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CSchulten- Job A#1

# ANNEX A

## Nuclear Criticality Safety

### 1. Scope of Nuclear Criticality Safety Assessment

1.1 A nuclear criticality safety analysis shall be performed for each light water reactor spent fuel storage facility system that involves the handling, transfer, or storage of spent fuel assemblies.

1.2 The nuclear criticality safety analysis shall demonstrate that each reactor spent fuel storage facility system is subcritical ($k_{eff}$ shall not exceed 0.95).

1.3 The nuclear criticality safety analysis shall include consideration of all credible normal and abnormal operating occurrences, including:

a) Accidental tipping or falling of a spent fuel assembly
b) Accidental tipping or falling of a storage rack during transfer
c) Misplacement of a spent fuel assembly
d) Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system.
e) Fuel drop accidents
f) Stuck fuel assembly/crane uplifting forces
g) Horizontal motion of fuel before complete removal from rack
h) Placing a fuel assembly along the outside of rack
i) Objects that may fall onto the stored spent fuel assemblies

1.4 At all locations in the reactor spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis shall demonstrate that criticality could <u>not</u> occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

1.5 The nuclear criticality safety analysis shall explicitly identify spent fuel assembly characteristics upon which subcriticality in the reactor spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis shall explicitly identify design limits upon which subcriticality depends that require physical verfication at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis shall explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

## 2. Calculational Methods and Codes

Methods used to calculate subcriticality shall be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety." (Endorses ANSI N16.9-1975)

## 3. Method to Establish Subcriticality

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks under normal and credible abnormal conditions shall be equal to or less than an established maximum allowable multiplication factor $k_a$; i.e.,

$$k_s \leq k_a \qquad \text{(Eq. 1)}$$

where

$k_s$ = the evaluated maximum multiplicaton factor of fuel in the spent fuel storage racks, including any necessary allowance for statistical uncertainties in the calculational technique such as in Monte Carlo calculations.

The maximum allowable multiplication factor shall be calculated from the expression:

$$k_a = k_c - \Delta k_u - \Delta k_m \qquad \text{(Eq. 2)}$$

where

$k_c$ = $k_{eff}$ computed for the most reactive fuel assembly at the most reactive point by the same calculational method which was used for the benchmark experiments.

<u>Note</u>: $k_c$ is the value of $k_{eff}$ that results from the calculation of the benchmark experiments using a particular calculational method. The value represents a combination of theoretical technique and numerical data. (For more detail, see Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety.")

$\Delta k_u$ = The uncertainty in the benchmark experiments.

$\Delta k_m$ = The value required to assure an accepted margin of subcriticality.

3.2 $\Delta k_u$ shall include both uncertainties in the benchmark experiments as well as uncertainties in the bias which result from extrapolation of the benchmark experiments into the range of parameters encountered in the spent fuel storage rack design.

3.3 $\Delta k_m$ shall provide an adequate margin of subcriticality under the operating limitations and Design Events I through IV, and shall be no less than 0.02 (new fuel when stored dry).*

3.4 In the absence of information that justifies a smaller margin of subcriticality, value of 0.05 shall be assumed for $\Delta k_m$ for the design of spent fuel storage racks (spent fuel).

## 4. Storage Rack Analysis Assumptions

4.1 [~~The-fuel-assembly-assumed-for-storage-facility-design-shall-be-one-of-the-following:~~] The spent fuel storage rack module design shall be based on one of the following assumptions for the fuel:

a) the most reactive fuel assembly to be stored at the most reactive point in the assembly life [~~with-no-allowance-for-fission-product-content-due-to-burn-up~~]; or

*Additions shown by underline and a vertical line in each margin. Deletions shown by brackets and crossouts.

b) the most reactive fuel assembly to be stored based on a minimum confirmed burn up. [~~if credit is taken for burnup, an allowable fuel assembly reactivity shall be established and it shall be shown by actual measurement that each fuel assembly meets this criterion before it is allowed to be placed in storage.~~] (See Annex B.)

Both types of rack modules may be present in the same storage pool.

4.2 Determination of the most reactive spent fuel assembly shall include consideration of the following parameters:

- maximum fissile fuel loading,
- fuel rod diameter,
- fuel rod cladding material and thickness,
- fuel pellet density,
- fuel rod pitch and total number of fuel rods within assembly,
- absence of fuel rods in certain locations, and
- burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design shall be the arrangement that results in the highest value of $k_s$ considering:

a) spacing between assemblies,
b) moderation between assemblies, and
c) fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of $k_s$ shall include consideration of the following:

a) eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,

b) dimensional tolerances,

c) construction materials,

d) fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e) presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f) presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Determination of burn up for storage shall be made in racks for which credit is taken for burn up. The following methods are acceptable:

a) a minimum allowed fuel assembly reactivity shall be established and a reactivity measurement shall be performed to assure that each assembly meets this criterion; or

b) a minimum fuel assembly burn up value shall be established as determined by initial fuel assembly enrichment or other correlative parameters and a measurement shall be performed to assure each fuel assembly meets the established criterion; or

c) a minimum fuel assembly burn up value shall be established as determined by initial fuel assembly enrichment or other correlative parameters and an analysis of each fuel assembly's exposure history shall be performed to determine its burn up. The analyses shall be performed under strict administrative control using approved written procedures. The procedures shall provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in Section 1.4.

The uncertainties in determining fuel assembly storage acceptance criteria shall be considered in establishing storage rack reactivity, and auditable records shall be kept of the method used to determine fuel assembly storage acceptance criterion for as long as the fuel assemblies are stored in the racks.

Consideration shall be given to the axial distribution of burn up in the fuel assembly and a limit shall be set on the length of the fuel assembly which is permitted to have a lower average burn up than the fuel assembly average.

5. Use of Neutron Absorbers in Storage Rack Design

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a) The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b) Fixed neutron absorbers shall be an integral, non-removable part of the storage rack.

c) When a fixed neutron absorber is used as the primary nuclear criticality safety control, there shall be provision to:

1) initially confirm absorber presence in the storage rack, and

2) periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water shall not normally be used in the evaluation of $k_s$. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

# ANNEX B

## Most Reactive Fuel Assembly to be Stored Based on a Minimum Confirmed Burnup

If credit is to be taken for fuel burnup in the design of spent fuel storage racks, an acceptable basis for setting and meeting the limit must be established. The rationale for this basis will evolve from many rather complex considerations.

Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, $^{235}U$ depletion, amount of burnable poison, plutonium buildin and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildin are not necessarily the same.

Consideration should be given to how burnup limits are selected and specified for a particular fuel type:

The allowable $^{235}U$ depletion in the spent fuels without burnable poison must not be set too high. If too much depletion is credited in the analysis compared to the range of $^{235}U$ depletion in spent fuel assemblies to be stored, the design could be nonconservative from the standpoint of criticality safety. On the other hand, if too little depletion is credited in the analysis compared to the spent fuel to be stored, then the design will be conservative. Thus a <u>maximum</u> depletion to be allowed in <u>design</u>

can be established consistent with the range of $^{235}U$ depletions expected in the spent fuel assemblies to be stored. (This limit would then correspond to the minimum depletion that would be allowed in a particular fuel assembly type destined to be stored in the racks.)

The allowable plutonium content in the spent fuel upon which design would be based must not be set too low. If design is based on too little plutonium compared to the range of plutonium concentrations that may be in the spent fuel assemblies to be stored in the racks, the design could be non-conservative from the standpoint of nuclear criticality safety. On the other hand, if too much plutonium is credited in the analysis of the storage racks compared to the spent fuel assemblies to be stored, then the design would be conservative. Thus, a minimum plutonium content to be allowed in design can be established consistent with the range of plutonium concentrations expected in the spent fuel assemblies to be stored.

(This limit would then correspond to the maximum plutonium content that would be allowed in a particular fuel assembly type destined to be stored in the racks.)

Credit for fission product content presents special problems, such as the identities and quantities of the various fission products present and how to evaluate the effect of decay rates on the credit taken. The allowable fission product content in the spent fuel upon which design would be based must not be set too high. If design is based on too high of a fission product content compared to the range of fission product concentrations that may be in the spent fuel assemblies to be stored in the racks, the design could be non-conservative from the standpoint of criticality safety. On the other hand, if too few fission products are credited in the analysis of the racks compared to the spent fuel assemblies to be stored, then the design would be conservative. Thus, with proper consideration a maximum fission product content to be allowed in design could be established consistent with the range of fission product concentrations expected in the spent fuel to be stored.

(This limit would then correspond to the minimum fission product content that would be allowed in a particular fuel assembly type to be stored in the racks.)

Finally, consideration should be given to the practical implementation of the spent fuel screening process. Factors to be considered in choosing the screening method should include: [~~Depletion of $^{235}$U and plutonium and fission product buildin cannot be easily or practically determined analytically. An obvious approach would be to translate the allowable burnup to a net allowable fuel assembly reactivity and then measure every fuel assembly to confirm that the minimum criterion is met.~~]

- accuracy of the method in determining the storage rack reactivity;
- reproducibility of the result, i.e., what is the confidence in the result?
- simplicity of the procedure; i.e., how much disturbance to other operations is involved?;
- accountability, i.e., ease and completeness of recordkeeping; and
- auditability.

# VALUE/IMPACT ASSESSMENT ON NUCLEAR POWER PLANT SPENT FUEL STORAGE FACILITY DESIGN

## 1. <u>PROPOSED ACTION</u>

### 1.1 <u>Description</u>

Each nuclear power plant has a spent fuel storage facility. General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

### 1.2 <u>Need for Proposed Action</u>

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested this guide be updated.

## 1.3 Value/Impact of Proposed Action

### 1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit application. Therefore, there should be a minimum of cases where the applicant and the staff radically disagree on the design criteria.

### 1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

### 1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

### 1.3.4 Public

No major impact on the public can be foreseen.

## 1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

# 2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also recommendations made by Task A-36 which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" would also be included.

## 3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

## 4. STATUTORY CONSIDERATIONS

### 4.1 NRC AUTHORITY

This guide would fall under the authority and safety requirements of the Atomic Energy Act of 1954, as amended. In particular under General Design Criterion 61, Appendix A, 10 CFR Part 50 of the NRC's implementing regulations.

### 4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by 10 CFR Part 51.5(a)(10) and does not require an environmental impact statement.

## 5. CONCLUSION

Regulatory Guide 1.13 should be updated.